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ABSTRACT

This handbook is designed to help users of radioactive materials to handle the radioactive material without exposing themselves or others to radiation doses in excess of maximum permissible limits. The discussion of radiation levels is in terms of readings from dosimeters and survey instruments. Safety in the handling of radioactive materials in research and practical situations where the quantity of materials handled is not great is stressed. Radioactive waste disposal is also discussed. (SL)

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National Bureau of Standards A. V. Astin, Director

Safe Handling of Radioactive Materials

Recommendations of the
National Committee on Radiation Protection

NCRP Report No. 30

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Foreword

This Handbook, prepared by the National Committee on Radiation Protection and Measurements, supersedes NBS Handbook 42, Safe Handling of Radioactive Isotopes, issued in 1949.

Since 1931, recommendations of the National Committee on Radiation Protection and Measurements (for many years known as the Advisory Committee on X-ray and Radium Protection and later as the National Committee on Radiation Protection) have been published as National Bureau of Standards Handbooks. The Bureau is pleased to have the continuing opportunity to increase the usefulness of these important reports by providing the publication outlet.

Since publication of Handbook 42, radionuclides have come into much wider use in research, medicine, industry, and agriculture. Their applications in radiography, level and thickness gaging, food processing, well logging, and radiotracer testing have introduced problems of radiation safety in many fields, but this diversified use has also broadened the base of experience in their safe handling.

More recent Handbooks (listed on the inside front cover) have expanded and kept up to date the principles of radiation protection first gathered together in Handbook 42, particularly with respect to internal exposure. The present publication sets forth these principles, making reference to other Handbooks where more detailed information may be obtained.

A. V. Astin, *Director*.

Preface

This Handbook is to be used particularly in conjunction with NBS Handbook 59, Permissible Dose from External Sources of Ionizing Radiation, and NBS Handbook 69, Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure. Handbook 59 sets limits on the exposure from external sources of radiation, whereas Handbook 69 sets limits on the body burden of radionuclides and the concentration in air and water consistent with these limits. The present Handbook is designed to help the user to handle radionuclides without exposing himself or others to doses in excess of these limits.

The accurate determination of radiation doses is a complex technical problem outside the scope of this Handbook. The availability of dosimeters and survey instruments from manufacturers and service companies is assumed as a basis for monitoring exposures, and the discussion of radiation levels is in terms of the readings from such instruments.

An attempt has been made to include in this Handbook the main considerations of safety in the handling of radionuclides gained from research and practical experience to date. The atomic product plants where uranium, plutonium, and by-product radionuclides are processed have the broadest experience in handling large amounts of radionuclides and have developed facilities and procedures which exemplify the maximum protection practices considered normally necessary in the field. The nuclear power industry should need no more, and radionuclides users in research laboratories, medical practice, and industry will need less. It is important for the practical utilization of radionuclides that the user be adequately protected; it is equally important that he not be hampered with unnecessarily elaborate and expensive safety procedures. Although the principles given are generally applicable, this Handbook has been prepared primarily for users of radionuclides in operations where the quantities of radioactive materials and the complexities of handling are not great.

The National Committee on Radiation Protection and Measurements (originally known as the Advisory Committee on X-ray and Radium Protection) was formed in 1929 upon the recommendation of the International Commission on Radiological Protection. The Committee consists of a Main Committee and twenty subcommittees. Each of the subcommittees is charged with the responsibility of preparing recommendations in its particular field. The reports of the subcommittees require approval by the Main Committee before publication.

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- Subcommittee 6. Handling of radioactive materials, J. W. Healy.
- Subcommittee 7. Monitoring methods and instruments, A. R. Keene.
- Subcommittee 8. Waste disposal and decontamination (this subcommittee has been inactivated.)
- Subcommittee 9. Protection against radiations from Ra, Co^{60}, and Cs^{137} encapsulated sources, C. B. Braestrup.
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- Subcommittee M-2. Standards and measurement of radiological exposure dose, H. O. Wyckoff.
- Subcommittee M-3. Standards and measurement of absorbed radiation dose, H. O. Wyckoff.
- Subcommittee M-4. Relative biological effectiveness, V. P. Bond.

The present handbook was prepared by the NCRP Subcommittee on Handling of Radioactive Materials with the following members:

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During the preparation of the initial drafts of this report, the Subcommittee Chairman was H. M. Parker. Particular assistance in editing of the last draft was given by D. E. Hull.

L. S. TAYLOR, *Chairman*
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Radiation Protection
and Measurements

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SAFE HANDLING OF RADIOACTIVE MATERIALS

1. General Considerations

1.1 Scope

Ionizing radiation from radioactive nuclides can, in sufficient quantity, cause observable changes in both living organisms and inanimate materials exposed to the radiation. When using radioactive materials or radiation, the user shall be responsible for minimizing the harmful effects.

Because the human senses cannot detect ionizing radiation, physical safeguards must be provided to protect workers and the public from harmful exposure and workers must be taught to use these safeguards to protect themselves.

Maximum permissible limits for occupational radiation exposure of workers, to both internal and external radiation, are set at levels sufficiently low that no appreciable bodily injury is expected to occur to the individual even during a lifetime of exposure. As used here, appreciable bodily injury means any bodily injury or effect that the average person would regard as being objectionable and/or competent medical authorities would regard as being deleterious to the health and well being of the individual. Exposure of people not occupationally involved is, in general, limited to much lower levels.

Radiation protection is particularly concerned with human beings, but protective measures must also be provided for plant and animal life beneficial to the human race, and for certain inanimate things, such as photographic and x-ray films and radiation instruments. In many cases, the measures taken to avoid contamination of experiments or to protect inanimate objects are more stringent than the measures required for protection of people. This handbook deals only with the protection of people.

Work with radionuclides varies from handling multicurie quantities of several hundred different isotopes in nuclear energy plants, down to handling millicurie or microcurie quantities of individual nuclides in hospitals or laboratories. The principles underlying the safe handling are the same in all cases, but the extent of control and the amount of complexity of equipment required vary greatly from one extreme to the other. In all cases, by safeguarding each radiation source, by proper use of controls and monitoring facilities, by adequate waste disposal, and by careful instruction of personnel the user of radionuclides can fulfill his responsibility to minimize the harmful effects of radiation.

Attention is directed to the usage of "shall" and "should" throughout the recommendations of the National Committee on Radiation Protection and Measurements. "Shall" is used when adherence to the recommendations is necessary to meet accepted standards of protection. "Should" indicates recommendations that are to be applied, where possible, in the interest of minimizing radiation exposure.

1.2 Available Radionuclides

At present, most radionuclides are obtained directly or indirectly from the U.S. Atomic Energy Commission, and the user must obtain a license from the Commission to possess such nuclides. Some states have assumed the regulatory function of the AEC, and when this is the case, application is made to the proper agency in those states. Licensees of the AEC are subject to radiation safety regulations published by the AEC in CFR 10 Part 20 [1],* which is supplied to applicants with the application forms. The user should be familiar with these regulations before applying for his license.

Secondary suppliers of radionuclides in various forms, such as labeled compounds or sterile solutions, are listed in the Isotope Index [2]. Suppliers of equipment for detection of radiation and services needed in connection with the use of radionuclide are listed annually in the November issue of Nucleonics [3]. A bibliography of procurement, regulations, uses, and training programs is published by the U. S. Atomic Energy Commission [4].

1.3 Radiation Protection Terminology

The terminology of radiation protection is explained in many textbooks, and glossaries are to be found in other Handbooks of this series [5–11], Radiation Hygiene Handbook [12], and the ASA Glossary [13]. Precise definitions for some of the quantities and units are given in appendix E. Terms and their explanations which are adequate for the purpose of this handbook are given below.[1]

---

* Figures in brackets indicate the literature reference at the end of this paper.

Absorbed Dose (D)—The energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest. A special[2] unit for this quantity is the rad.

Activity (A)—The number of disintegrations of a quantity of radionuclide per unit time. Care must be exercised in a measurement of activity to insure that the time is short enough not to be influenced by the radioactive decay and that the number of disintegrations is large enough to be statistically significant.

Alpha Particles, Alpha Rays—Particulate ionizing radiation consisting of helium nuclei traveling at high speed.

Background Radiation—Radiation arising from sources other than the one directly under consideration. Background radiation due to cosmic rays and natural radioactivity is always present. There may also be additional background radiation due to the presence of sources of radiation in other parts of the building and/or area.

Beta Particles, Beta Rays—Particulate ionizing radiation consisting of electrons or positrons traveling at high speed.

Contamination (Radioactive)—Deposition or presence of radioactive material in any place where it is not desired, and particularly in any place where its presence can be harmful. The harm may be in vitiating the validity of an experiment or a procedure, or in being a source of danger to persons.

Controlled Area—A defined area in which the occupational exposure of personnel to radiation or to radioactive material is under the supervision of a radiation safety officer. (This implies that a controlled area is one that requires control of access, occupancy, and working conditions for radiation protection purposes.)

Critical Organ—That part of the body that is most susceptible to radiation damage under the specific conditions considered.

Curie (c)—The special unit of activity; $1c=3.7\times10^{10}$ disintegrations/sec.[3]

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[1] Explanations given in this section are derived from and are meant to have the same meaning as the more precise definitions in appendix E. These explanations are thought to be generally adequate but in case of question the definitions in appendix E are to be considered the primary source.

[2] See appendix E.

[3] See footnote 6 p. ??.

Daughter—An isotope formed by the decay of a given radioactive isotope. The daughter may be either radioactive or stable.

Distribution Factor (DF)—A factor used to express the modification of biological effect due to non-uniform distribution in the body of the radionuclides in question.

Dose Equivalent (DE)[4]—A concept used in radiation-protection work to permit the summation of doses from radiations having varying linear energy transfers, distributions of dose, etc. It is equal numerically to the product of absorbed dose in rads and arbitrarily defined quality factors, dose distribution factors and other necessary modifying factors. In the case of mixed radiations, the dose equivalent is assumed to be equal to the sum of the products of the absorbed dose of each radiation and its factors.

Exposure—A measure of x and gamma radiation at a point. Its special[5] unit is the roentgen.

Exposure Rate, Absorbed Dose Rate—The time rate at which an exposure or absorbed dose occurs; that is, exposure or absorbed dose per unit time. It implies a uniform or short-term average rate, unless expressly qualified (e.g., peak dose rate). In protection work it is usually expressed in mR/hr, mrads/hr.[6]

Gamma Radiation, Gamma Rays—Electromagnetic radiation of short wavelength and correspondingly high frequency, emitted by nuclei in the course of radioactive decay.

Quality Factor (QF)—The linear-energy-transfer-dependent factor by which the absorbed dose is to be multiplied to obtain, for purposes of radiation protection, a quantity that expresses on a common scale for all ionizing radiations the irradiation incurred by exposed persons.

Rad (rad, rads)—The special[7] unit of absorbed dose, 100 ergs per gram of irradiated material at the place of interest.

Rem (rem, rems)—The special[8] unit of dose equivalent.

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[4] This term is similar to what was formerly called RBE dose but is reserved for human radiation protection work. It was coined to make clear the distinction between the problems involved in such work and those involved in work with experimentally determined values of RBE.

[5] See appendix E.

[6] Dose rates are expressed in rads or rems per hour or multiples of submultiples of such units, e.g., rads per second, millirads per hour (mrads/hr). Similarly, activity may be expressed in curies or multiples thereof.

[7] See appendix E.

[8] See appendix E.

Roentgen **(R)**—The special[9] unit of exposure equal to the production in air of ions bearing 2.58×10^{-4} coulombs of charge of either sign by electrons generated per kilogram in air.

Specific Activity Nuclide—Total activity of a given nuclide per gram of the radioactive nuclide (e.g., *c* plutonium^{239}/gm plutonium^{239}).

X Radiation, X Rays—Electromagnetic ionizing radiation originating outside the atomic nucleus. X rays are indistinguishable from gamma rays of the same energy, the distinction being one of source.

1.4 Hazards in Handling Radionuclides

The potential health hazards of handling radioactive materials generally arise from the radiations which they emit. Protection must be provided against both external radiation during handling and intake of the radionuclide into the body with resulting internal radiation.

Excellent summaries of the effects of radiation on humans and the levels of exposure to radiation sources now generally prevalent are available in the recent reports of several scientific committees [14–16].

a. External Radiation Sources

Exposure of the body to penetrating x, gamma, or neutron radiation affects different cells, tissues, and organs in varying degrees. In particular, the blood-forming organs, the gonads, and the lens of the eye, are believed to be more sensitive than other tissues. The effect on a given organ depends on the dose, the type of radiation, the penetrating power of the radiation, and the time period over which the exposure is given. To take account of this, limits are established for exposure of the critical organs. Concurrent internal exposure is considered to be additive to external exposure.

External beta or soft x radiation penetrates only a few millimeters into the superficial layers of tissue, and very little if any reaches the major blood-forming organs. The permissible dose for skin of the whole body is higher than for underlying blood-forming organs reached by penetrating radiation [20]. The outermost layer of skin is a dead cornified layer which acts as a filter and the dose is computed for the regenerative zone immediately beneath. In general, the filter thickness is taken as 7 mg/cm^2. For the palm of the hand, where the dead layer is thicker, a value of 40 mg/cm^2 is often used.

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[9] See appendix E.

Surface exposure of skin to alpha radiation from radionuclides when the radioactive material is not absorbed through the skin or admitted through a wound is of no concern because the alpha particles are completely absorbed in the outer layer of dead cornified skin without harm to the underlying living tissue.

Exposure of the hands and forearms and feet and ankles to penetrating x, gamma, or neutron radiation is permitted at a higher level than whole-body exposure because of the absence of major blood-forming organs in these extremities. The lens of the eye is considered radiosensitive because of the possibility of radiation induced cataract; limits similar to those for whole-body exposure are recommended for the lens.

Maximum permissible limits for exposure of people to external radiation are given in NBS Handbook 59 [7].

b. Internal Radiation Sources

Deposition of radionuclides in the human body may result from ingestion, inhalation, or absorption through the intact or injured body surface. Exposures may be either acute or chronic. Acute exposure implies single or a few closely spaced exposures, while chronic exposure involves individual exposures continuing over a long period of time. The body organ or tissue receiving the radiation exposure which results in the greatest damage to the body is designated as the critical organ.

During the initial period following ingestion of radioactive materials (up to 30 hours for the standard man), that portion not absorbed will irradiate the walls of the gastrointestinal tract and may result in making it the critical organ. After elimination of radioactive material from the gastrointestinal tract, the radionuclides absorbed into the body and deposited in other organs (i.e., strontium in the bone, iodine in the thyroid, etc.) result in a radiation dose to these organs. In continuous intake, the over-all exposure results from nuclides in the gastrointestinal tract as well as from those deposited in other organs. A review of these factors and the calculation of maximum permissible limits from known data are given in NBS Handbook 69 [17] as well as in the report of the ICRP Subcommittee on Internal Emitters [18].

During inhalation, materials soluble in the body fluids will be rapidly absorbed into the bloodstream and deposited in other organs of the body. Particles of insoluble materials greater than about ten microns in diameter will be mostly captured in the nasal and bronchial passages and will irradiate adjacent tissues until eliminated by ciliary action or other mechanisms. A portion of the smaller particles will penetrate to the alveoli of the lung where they may be retained for days or months. The portion of material in the respiratory tract which is eliminated by ciliary action may be swallowed with consequent irradiation of the GI tract and possible absorption in the body. A detailed discussion of these mechanisms may be found in the report of the Inhalation Hazards Subcommittee of the Committee on Pathological Effects of Atomic Radiations [19] as well as in the reports of the Internal Dose Subcommittee of both the NCRP and the ICRP [17, 18].

Absorption of radioactive materials, through an open wound or even through intact skin, is a potential hazard when more than tracer doses are handled. Retention of radionuclides in the skin itself can be a hazard [20].

c. Exposure to Nonhuman Systems

Although most considerations in radiation protection are directed toward humans, consideration must be given to possible damage in other things important to man. Distribution of nuclides in the ecological environment limits the disposal of waste both because of possible damage to organisms of economic importance [21, 22] and because of possible contamination of food supplies [23, 24]. This is an exceedingly complex problem because of differences in uptake by plants and animals in different environments. In many cases deposition on crops or uptake by crops or organisms used as food supplies for humans require limits for radionuclides in air or water several orders of magnitude more restrictive than direct human consumption of the air or water [21].

X-ray film is sensitive to stray radiation or radioactive contamination in the film or packing materials. Other photographic films and papers are generally less sensitive but can be fogged by excessive exposure.

Low-level measurements of radioactivity are handicapped by background radiation and by contamination of instruments. While stray radiation can frequently be eliminated by shielding, contamination of instruments or samples can be insidious and more difficult to detect and avoid. In many laboratories, the prevention of such "cross-contamination" provides a more restrictive limit than protection of personnel. Contamination in materials used for constructing low-level instrumentation is increasing and will require increased diligence in the future to preserve present backgrounds.

1.5 Principles of Radiation Protection

Exposure to radioactive materials above the maximum permissible levels can be avoided by proven protective measures in the handling of radionuclides.

The fundamental objectives of such protective measures are:

1. To maintain exposure to external radiation as low as feasible.
2. To minimize entry of radionuclides into the human body by ingestion, inhalation, absorption, or through open wounds when unconfined radioactive material is handled.

To accomplish these objectives requires positive planning and diligent execution of procedures, beyond the usual care taken in work with other materials. It is necessary to analyze in advance the hazards of each job; to provide safeguards against foreseeable accidents; and to use protective devices and planned emergency procedures in accidents that do happen.

2. Control of Radiation Exposure

2.1 Exposure Limits

On April 15, 1958, the National Committee on Radiation Protection and Measurements issued a statement setting forth revised recommendations on maximum permissible radiation exposures to man. That statement also introduced the concept of accumulated dose limitation and revised some of the control limits [25]. This statement is attached as an addendum to Handbook 59 [7], and the dose limits it sets forth supersede those in the body of the Handbook. It is essential for anyone handling radionuclides to be familiar with the limits specified by the NCRP, and it is assumed in what follows in this Handbook that the reader also has NBS Handbook 59 for ready reference. This chapter deals with methods of assessing the radiological hazards of handling nuclides, and means for controlling the exposures within the limits set in Handbook 59.

2.2 Control of External Irradiation

The three factors which determine radiation exposure from a given radioactive source are *distance, time,* and *shielding.*

Protection against external exposure from alpha rays is not needed and to a large extent protection for beta rays is easily provided by shielding; but radiation safety from the more penetrating gamma rays is most conveniently and economically attained by consideration of these three factors in the order given.

Increasing distance from the source is frequently the most effective and the most economical means to reduce radiation exposure. The radiation level varies inversely with the square of the distance. Application of this principle dictates the use of tongs or other long-handled tools for handling radionuclide preparations emitting significant levels of radiation. They should never be picked up with the fingers; even short forceps provide a large attenuation of the radiation dose from that given to the skin in direct contact. It also means that radionuclide storage should be removed from the immediate work area.

Decreasing the time of exposure decreases the dose in direct proportion. This emphasizes the importance of advance planning of each step of a procedure, and the rehearsal of key operations with inactive materials.

When maximum distance and minimum time do not insure an acceptably low exposure, the gamma-radiation source must be shielded. Gamma-rays are attenuated according to an exponential law, typically (e.g., in the case of Co^{60} or radium) by a factor of ten by approximately 1½ in. of lead or 1½ feet of water.

In planning any operation which will expose personnel to gamma-rays, the potential exposure should be estimated beforehand. A formula useful for this purpose is:

Radiation exposure = (gamma-ray constant × Attenuation × Time) ÷ $(\text{Distance})^2$.

In symbols:

$$D = \frac{SAT}{d^2}$$

To use this formula determine S, the gamma ray constant in mR/hr at 1 meter, by multiplying the number of millicuries by the specific gamma ray constant of the nuclide given in table 1 or by measuring the source with a survey meter. Divide S by the square of *d*, the distance in meters at which the material will be handled. This is the radiation level, in mR/hr. Multiply this by *T*, the time that the operation will require. The result, *D*, is the exposure in mR which is expected from the planned operation with the material unshielded ($A=1$). If this is less than the permissible limit (say 20 mR/day for routine work, or 100 mR for a job that may be repeated as often as once a week), then the operation may be carried out without undue hazard, if other exposures are sufficiently low.

If the value of *D* is above the permissible level, it can be reduced by increasing *d* (longer-handled tools) or by decreasing *T* (rehearsal of operation). When this cannot be done, the thickness of shielding required to reduce *A* to a value which will give a permissible value of *D* can be calculated from the half-value thicknesses given in table *1*.

To illustrate the order of magnitude of radiation exposures for typical jobs, consider the problem of moving 1 curie of cobalt-60 from a shielded shipping box to a shielded hood with a 6-ft pair of tongs. From table *1* we find that the source strength is $1000 \times 1.33 = 1330$ mR/hr, at 1 meter. At 2 meters the radiation level is $1330/(2)^2 = 330$ mR/hr, or 6 mR/min. If the operation can be completed in 30 seconds, the exposure would be $6 \times \frac{1}{2} = 3$ mR, and no shield would be needed.

Again, consider the potential exposure of the chemist who plans a chemical preparation with 50 mc of cobalt-60, during which he may be standing at the hood, 50 cm from the source, for intervals adding up to 30 min. The exposure from the unshielded material would be $50 \times 1.33 \times \frac{1}{2}/(0.5)^2 = 133$ mR. It would be advisable to shield the

TABLE 1. *Specific gamma-ray constants and half value thicknesses for various nuclides*

Nuclide	Energy Mev	Specific gamma ray constant mr/hr/mc at 1 meter	Half-value thickness in.			
			lead	iron	water	concrete
K^{42}	1.53	[b]0.15	0.46	0.72	4.7	2.3
Sc^{46}	[a]1.00 (avg)	1.10	.35	.58	3.8	1.8
Cr^{51}	.32	[b]0.018	.077	.34	2.3	1.1
Fe^{59}	[a]1.20 (avg)	.63	.41	.64	4.2	2.0
Co^{60}	[a]1.25 (avg)	1.33	.42	.65	4.2	2.1
Zn^{65}	1.11	.30	.39	.61	4.0	1.9
Rb^{86}	1.08	[b] .058	.38	.60	3.9	1.9
Cs-Ba^{137}	.67	.31	.24	.48	3.1	1.5
Ba-La^{140}	1.60	1.40	.48	.73	4.7	2.3
Au^{198}	.41	.23	.12	.38	2.6	1.2

[a] Average.

[b] Gamma-rays emitted in a small fraction of the disintegrations.

material in this operation. If 2-in. lead bricks are used, the attenuation would be $2^{-(2/0.42)}=0.04$ and the exposure 5 mR, an acceptable figure. In more precise calculations, buildup of scattered gamma radiation must be taken into account (see 4.1.*c* and appendix B).

2.3 Control of Internal Irradiation

Permissible limits for internal exposure have been derived by the National Committee on Radiation Protection and Measurements for about one-quarter of the known radioinuclides [17]. Tables such as table 2 have been found useful as guides to the degree of protection required for handling various quantities of the more common radioactive nuclides.

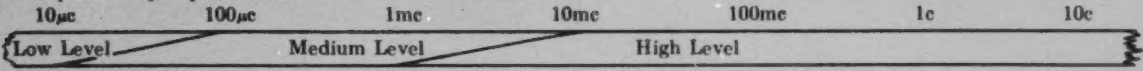
The estimated relative hazards of radionuclides are based on their physical properties and their maximum permissible concentrations in air and water. The classifications in table 2 are for the *soluble* forms, primarily, of the listed radionuclides which may be deposited internally in humans. They take into account, where possible, the types of compounds in which such nuclides are usually encountered, their specific activity, volatility, and the maximum permissible limits. Such brief classifications include numerous arbitrary placements, resulting from generalization of specific variables.

In each hazard group the radionuclides are arranged in order of ascending atomic weight. An asterisk (*) denotes a radionuclide which emits gamma radiation in amounts significant for external exposure. The levels refer to the activities of radionuclides that may be present (as shown by the upper scale) and to the degree of protection required against internal deposition. Medical or research laboratories and industrial installations should have ventilation and safeguards against contamination corresponding to such classifications, as discussed in sections 4 and 5.

The slant boundaries indicate borderline zones in which the adjacent quantity of a *particular* radionuclide may be too much or too little for the indicated degree of protection, but especially emphasize that there are no sharp transitions between the quantity levels, group classifications, or associated protection techniques. Modifying factors may, and in the more hazardous cases should, be applied to the quantities handled, according to the complexity of the handling procedure. For normal chemical operations with the "average" radionuclide in each group, the modifying factor may be taken as one. For very

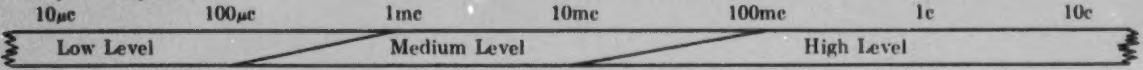
TABLE 2. *Hazard from absorption into the body*

Group 1. Very High Hazard.


10μc
100μc
1mc
10mc
100mc
1c
10c
Low Level
Medium Level
High Level

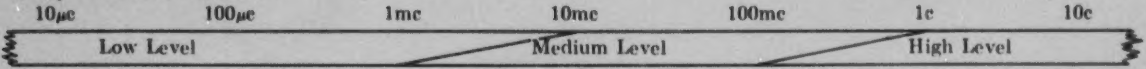
*Pb^{210}, Po^{210}, *Ra^{226}, *RA^{228}, Ac^{227}, Th^{228}, Th^{230}, Np^{237}, Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}, *Am^{241}, Cm^{242}

Group 2. High Hazard.


10μc
100μc
1mc
10mc
100mc
1c
10c
Low Level
Medium Level
High Level

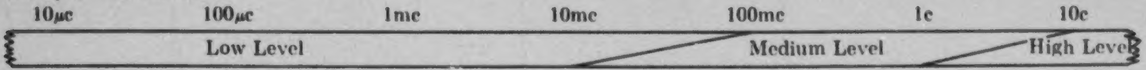
*Na^{22}, Ca^{45}, *Sc^{46}, *Co^{60}, Sr^{90}, *Ru^{106}, I^{129}, *I^{131}, *Cs^{137}, *Ce^{144}, *Eu^{154}, *Ta^{182}, Bi^{210}, At^{211}, Ra^{224}, U^{233}

Group 3. Medium Hazard.


10μc
100μc
1mc
10mc
100mc
1c
10c
Low Level
Medium Level
High Level

C^{14}, *Na^{24}, Si^{31}, P^{32}, S^{35}, Cl^{36}, *K^{42}, Sc^{47}, *V^{48}, *Cr^{51}, *Mn^{54}, *Mn^{56}, Fe^{55}, *Fe^{59}, *Cu^{64}, *Zn^{65}, *Ga^{72}, *As^{76}, *Rb^{86}, Sr^{89}, Y^{90}, Y^{91}, *Zr^{95}, *Nb^{95}, *Mo^{99}, *Ru^{103}, *Rh^{105}, Pd^{103}, Ag^{105}, Ag^{111}, *Cd^{109}, *Sn^{113}, *Te^{127}, *Te^{129M}, *Ba^{140}, *La^{140}, Pr^{143}, Pm^{147}, Sm^{151}, *Ho^{166}, *Tm^{170}, *Lu^{177}. *Re^{183}, *Ir^{190}, Ir^{192}, *Pt^{191}, *Pt^{193}, *Au^{196}, *Au^{198}, *Au^{199}, *Tl^{200}, *Tl^{201}, Tl^{202}, Tl^{204}, *Pb^{203}, Rn^{220}, *Rn^{222}, U^{235}

Group 4. Low Hazard.


10μc
100μc
1mc
10mc
100mc
1c
10c
Low Level
Medium Level
High Level

H^{3}, *Be^{7}, C^{14}, F^{18}, Ni^{59}, Zn^{69}, Ge^{71}, U^{238}, Natural Thorium, Natural Uranium, Noble Gases.

*Emits gamma radiation in significant amounts.

simple wet operations, up to ten times the indicated quantity of radionuclide may be handled without appreciable change in degree of protection. For storage of stable materials, such as stock solutions and most solids, up to 100 times the quantity indicated in table 2 may be handled with minimal extra precautions. On the other hand, complex wet operations with risk of serious spills may require limiting the quantity to as little as one-tenth, and dry and dusty operations to as little as one-hundredth, unless protection facilities and procedures are considerably improved.

The maximum permissible amounts of radionuclides in the human body and the maximum permissible concentrations in air and water for direct human consumption are listed and explained in National Bureau of Standards Handbook 69 [17]. These limits are based on the quantities of radionuclides which deposit in specific critical organs and the resulting radiation dose. The values have been specifically calculated for air which is inhaled and for water which is drunk by the human.

In cases where radionuclides are regularly discharged to the environment, the actual limitation on the quantities of radionuclides which can be permitted in the air or water may be lower, depending on the concentration of these nuclides resulting from natural deposition processes or from biological accumulation [21, 22]. In such cases, monitoring of the environment should include measurements of samples to assure that such accumulation of the radioactive material by biological materials will not result in over-exposure to man from ingestion of food derived from these sources. If no food is taken from the contaminated area, the damage to the fish or water fowl may be the limiting factor in the concentrations permitted in the streams. Determining the concentrations in food and water and the quantity of these materials actually ingested will permit effective control of the biological accumulation and transfer to man.

The determination of radiation exposure from a mixture of internal emitters becomes more complicated when the radiation comes from nuclides of several elements because their diverse chemistry leads to differences in location of deposition in the body and the radiations have different penetrations. In order to arrive at the total dose rate from a mixture of radionuclides, the contribution of each to the dose rate must be determined and summed for each organ significantly affected. The maximum permissible concentration of the mixture is, then, the total concentration that leads to the MPD in one organ or another. The maximum permissible concentrations of radionuclides in a mixture can be estimated from the following formula

$$\sum\left(\frac{C_i}{\mathrm{MPC}_i}\right)\leq 1$$

[26]: in which C_i is the concentration of the ith radionuclide and MPC_i is its maximum permissible concentration.

In some cases, not all of the radionuclides in a mixture will be known. In such cases, the maximum permissible limits used for the unknown fraction of the total radioactive materials should be the most conservative of those not identified. Choice of the maximum permissible limit to be used can be based on the values presented in Handbook 69 [17] for radionuclides emitting the type of radiation involved.

2.4 Irradiation by Mixed Radiations

The biological effect of irradiation by more than one type of radiation (i.e., gamma, beta and alpha) is obtained by summing the effects of the various radiations. Heavy particles ionize more densely than electrons and their biological effect per ion pair produced is usually greater. Thus, one rad of such radiation is usually equivalent to several rems.

The Quality Factor (QF) takes account of the LET or quality of the radiation in question. For most radiation protection purposes the dependence of the effect upon the LET or quality of the radiation is most marked. Other factors such as rate dependence and distribution of the radionuclide in the organ may be important in some applications. Since maximum permissible limits are based upon low level, chronic exposures, a series of values for the quality factor based upon late effects has been adopted. Thus, mixed radiation doses, computed in rems using the relationships given in table 3 are considered to be additive.

If several sources of radiation (e.g., external and internal) affect the same organ systems, they must be added as fractions of their respective maximum permissible limit and the situation managed so that the sum does not exceed the maximum permissible limit for that organ system.

Table 3.—*Simplified relationships for adding exposures to mixed radiations*

Radiation	Limiting criterion	Relationships
X rays, gamma rays, and electrons and β rays of all energies.	Whole-body irradiation (blood-forming organs critical).	1 rad = 1 rem[a]
Fast neutrons and protons up to 10 Mev.	Whole-body irradiation (cataract-formation).	1 rad = 10 rem
Alpha particles	Carcinogenesis	1 rad = 10 rem[b]
Heavy recoil nuclei	Cataract-formation	1 rad = 20 rem

[a] For certain specific radionuclides emitting low energy beta particles such as tritium and for x rays up to about 30 kvp, the value 1 rad = 1.7 rem is used [16].

[b] For elements which are fixed in bone, a relation may also be obtained by comparing the ionization with that produced by 0.1 μc of Ra^{226} taking into account differences in the pattern of distribution.

3. Personnel

3.1 Supervision

The supervisor of the laboratory has the responsibility to protect both the workers and the general public from the nuclides being used. He should be familiar with the basic principles of protection from radiation and radioactive materials in order to properly discharge this responsibility, although for details he may consult with the specialists who are more familiar with the over-all requirements. He shall see that the work of the group is properly planned, in detail commensurate with the degree of hazard as indicated in table 2, before operations are started. He should see that instructions for standard procedures are available for repetitive jobs and that special detailed procedures are prepared prior to any special work involving hazardous quantities of radioactive materials. These procedures shall include the principal steps to be taken in the event of an accident involving these materials. He shall prepare and revise as needed rules regarding the handling of food in the laboratory, use of protective equipment, wearing of personnel meters, personnel monitoring, storage of radioactive materials, etc. He shall see that these rules are enforced. He shall acquaint each and every worker with, and shall train him to follow these procedures, as necessary for the worker's own protection and the protection of others.

3.2 Radiation Protection

Wherever radioactive materials are used, there shall be a competent person responsible for the radiation protection program. His duties will vary depending upon the type of organization in which the work is being carried out, but he shall assist the supervisor in carrying out his program of radiation safety by advising on all phases of the work in matters of radiation protection. This person should preferably be in a position in the organization where he can effectively consult with those establishing the policy for the laboratory group or the procedure for carrying out work. His duties shall include instructing and training the workers in the safe handling of radionuclides; instructing workers and visitors in the use of protective equipment and procedures; arranging for proper waste disposal; keeping suitable records on personnel exposures; investigating accidents or unusual occurrences; seeing that properly calibrated instruments are available for workers or persons doing monitoring; and carrying out monitoring in cases where unusual contamination is suspected.

Radiation protection personnel must have training and experience in the field commensurate with the complexity of work. They must know the types of hazards involved with the various kinds of nuclides handled. They must be familiar with the maximum permissible exposures and concentrations of nuclides, and the factors which determine them. They must know the necessary techniques for monitoring and how to interpret the results of such monitoring.

In small groups handling one or a few radionuclides the radiation protection expert may be one of the working group with authority and responsibility for radiological safety. In larger organizations with staffs of more than about 25 people engaged in work with medium or high level quantities of a variety of radionuclides one or more full-time personnel qualified in radiation protection may be required.

Courses in the techniques required for using radionuclides and in the safety considerations in handling are offered by several groups, including the Oak Ridge Institute of Nuclear Studies and by the U.S. Public Health Service at the Robert A. Taft Sanitary Engineering Center at Cincinnati, Ohio. Other short courses are becoming available at various universities. It is often possible to begin useful work with radionuclides by taking advantage of these courses for training a person who may have no previous experience with radioactivity but who is otherwise technically competent.

3.3 Selection and Instruction of Workers

Persons who are neat and careful are preferred as radionuclide workers, since they are less likely to be involved in accidents or spread of contamination. In handling radionuclides, skill in radiation protection is as necessary as skill in physical, chemical, biological, or industrial manipulations. Management should recognize "accident-proneness" and persons failing to develop such skills and who have a bad record of involvement in contamination incidents or over-exposure should be advised to transfer to other occupations. Persons who have a history of sudden physical seizures, such as fainting, should not be permitted to work with high levels of radionuclides. At the time a person is employed for radiation work, his occupational radiation exposure records should be obtained from his previous employer.

All persons employed in radiation work shall be informed, in detail, of the materials which they are to handle and of the possible hazards connected with the work. They shall be instructed regarding local rules and provisions for protection and shall be expected to observe these rules and employ these provisions in all details. The employee shall be instructed in the use of protective equipment and in the use and interpretation of the readings from monitoring instruments. He shall report any injury or unusual incident so that the possibility of overexposure or internal deposition can be investigated. The person responsible for the work shall see that it is planned in detail before execution. The detail required here is dependent upon the quantities and hazards of the radioactive material.

It is important that workers with radionuclides be considered as potentially exposed for the remainder of their lifetimes. It shall not be assumed that they will work with radiation for only a few years.

3.4 Personal Cleanliness

Radionuclides can be taken into the body by a number of routes including ingestion or absorption through the skin. Extreme personal cleanliness and care are therefore needed. Different kinds of work with radionuclides will have different degrees of contamination hazard. In work with sealed sources, contamination will be a very minor problem, barring rupture of the container and dispersion of the source material. Laboratories using only low-level quantities (table 2) of a few nuclides will have a small, but finite, risk of contamination of workers and work areas. In large installations where high-level quantities are handled, continuous vigilance will be required to minimize the escape of radionuclides to the work area. The skill, conscientiousness, and number of personnel all have a bearing on the facilities and procedures that must be provided to control contamination. To maintain personal cleanliness, the following measures in varying degree have been found necessary.

Hands should be washed frequently, and shall be washed before eating, smoking, and at the end of each work period. No edibles of any kind—food, gum, candy, beverages—shall be brought into contaminated areas or areas that may become contaminated between radiation control surveys. Smoking shall be prohibited in such zones, unless written approval of persons responsible for radiation protection has been obtained, since radioactive contamination may be transferred to the lips or inhaled as the contamination is burned. Personnel should refrain from using personal items, i.e., pocket knives, handkerchiefs, lipsticks, etc., in the work area. Persons with open wounds on the hands, or other portions of the body which may come into contact with radioactive materials, should not handle medium or high levels of radionuclides without an adequate waterproof covering on the wound. Pipetting of solutions containing radioactive materials shall be done with suction devices. Mouth pipetting shall be strictly forbidden.

The hands, other uncovered skin, and clothing should be surveyed frequently during the work with instruments sensitive to the radiations involved, and if any contamination is found, immediate steps shall be taken to remove it. Surveys shall be made before leaving a contaminated zone, at which point protective clothing should be discarded.

Where contamination is frequently encountered, wash basins should have a foot- or knee-operated valve. Emergency showers as well as protective devices such as face shields, respiratory devices, etc., may be needed near each work area where liquid or powdered radioactive material is handled. Information on decontamination of personnel is given in section 5.4 and on control of contamination in section 5.5.

Personnel should keep their work areas free from equipment and materials not needed for the immediate work. Orderliness is a prime requirement for eliminating the spread of contamination. After use, equipment should be decontaminated or stored in a controlled location. Radioactive materials should be safely disposed of or returned to storage as soon as no longer needed.

3.5 Medical Examinations

Except in the case of exposure to radiation above the maximum permissible limits, no special medical examinations other than those considered good industrial or medical practice should be required. Clinical examinations should not be relied upon as indications of poor working conditions since positive indications on these tests are evidence of gross overexposure. Preemployment physical examination is always advisable to reveal any physical conditions that later may be attributed to radiation exposure. The preemployment examination should include medical history, radiation exposure history, physical examination, and, at the discretion of the medical officer in charge, a complete blood count.

In large organizations, the staff may include medical specialists who are available in case of radiation accidents; otherwise, arrangements should be made beforehand for the services of a physician who has familiarized himself with the diagnosis and treatment of radiation injuries and radioisotope poisoning.

The National Committee on Radiation Protection and Measurements makes the following statement regarding blood counts [27]:

1. Provided that radiation monitoring of personnel, and where applicable, of sites, is carried out by instruments (film badges, pocket meters, etc.) in all circumstances involving potential exposure to penetrating ionizing radiation, blood counts should no longer be required as a *method of monitoring*.
2. Blood counts as a part of preemployment, interval and terminal examinations are good medical practice—to be done at the discretion of the medical officer in charge—but not as a part of a monitoring service.
3. Blood counts are a necessary part of the medical examination of anyone overexposed to penetrating ionizing radiations.

3.6 Urinalysis and Other Tests

Special tests for determining the presence of internal emitters in the body are desirable for persons handling intermediate or high-level quantities (table 2) of unconfined radioactive materials. These tests consist of analyses of excreta, analyses of biopsy samples, and in recent years, the measurement of the radiation from the total body. In radiation protection practice, the term "bioassay" has been adopted for measurements of radioactive materials in blood, urine, feces, skin, and other body tissues or fluids. Scheduling of such tests depends upon the exposure history of the individual, past history with the work group, and the nature of possible accidents in which radioactive material could have been taken into the body. The radioactive half-life and the rate of biological elimination of the nuclide should be considered in timing the collection of samples for bioassay in relation to the times of potential exposure.

It is not expected that laboratories and industries using small quantities of radionuclides will be equipped and staffed to perform such bioassays and interpret the results. There are companies throughout the country specializing in such analyses[3].

a. Urine and Blood Samples

The analysis of urine is the most frequently used indicator of quantity of radioactive material in the body. In obtaining samples of this nature, care should be taken to insure that extraneous contamination is not introduced from the hands or clothing of the individual being sampled. Most biological samples contain natural radiopotassium in amounts which may mask the added radionuclides for which tests are being made. Either the potassium can be chemically separated from the sample or the radionuclide of concern can be separated for measurement. In general, it is advisable to analyze for the specific nuclide of interest so that the results may be more easily interpreted. Urine analysis is especially recommended for monitoring exposures to tritium. Analytical procedures are available for many of the common nuclides including tritium, strontium, cesium, radium, uranium, thorium, and plutonium [28-31].

The sensitivity of the analytical procedure should be adequate to detect the small amounts eliminated at body burdens well below the maximum permissible values [17]. It should be remembered that chronic exposure can result in the slow buildup of radioactive materials in the body with indications in the urine at any given time considerably lower than would be expected from the same total intake on an acute basis [32, 33]. In general, when the over-all elimination rate is rapid, the sampling intervals should be more frequent. However, for the most rapidly eliminated radionuclides, work control procedures may be established on the basis of daily to weekly bioassays to make better use of such data.

In the event of suspected exposure to an acute intake of radioactive material, prompt bioassy of urine samples will permit more accurate estimation of the immediate and long-term exposures, and will guide medical treatment. Frequently in such accidents, analyses of blood as well as urine are of considerable use [33]. The interpretation of results obtained by this technique is difficult, and at this time, uncertain. An estimation of body burden is usually made by reference to human data wherein excretion rates have been measured following administration of a known quantity of the radionuclide or element [32, 33]. The use of a single half-life to characterize the rate of biological elimination is not satisfactory for interpretation purposes except for a few selected nuclides such as tritium or sodium where the material is not tightly bound into the body. The results of urine and blood samples represent the radioactive material which has been taken into the blood stream for further movement through the body. They do not represent material in the lung or the GI tract which has not been absorbed into the bloodstream. The present methods of interpretation [33] of these data provide only semiquantitative indications of the quantity of radioactive material, but are extremely valuable in providing indications of work conditions which should be corrected.

b. Feces Samples

Feces assays are valuable when raioactive materials not readily soluble in body fluids are taken into the body by inhalation [19, 32, 33] or ingestion. Due to the deposition of radioactive material in the upper nasal and pulmonary passages, a fraction of the inhaled material will be removed to the throat by ciliary action and swallowed. Nasal smears, sputum, and irrigations of the nasal passages will frequently give indications of the material deposited in this fashion. Depending upon the particle size of the material inhaled, it is possible to have very high-level feces samples or nasal smears without a corresponding deposition of radioactive material in the body. In general, the results of such tests are not interpretable in terms of the quantity of material remaining in the body following this initial clearance. They do indicate, however, that undesirable atmospheric conditions have been encountered. There are no good methods of determining the lung burden (i.e., material not yet absorbed from the lung passages) from excreta analyses.

c. Expired Air Samples

Special techniques have been or may be developed for specific radionuclides. It is possible that analysis of a short-lived daughter which is poorly retained in the body, may provide a good indication of the quantity of a long-lived parent which is tenaciously retained. For example, radium-226 deposition is determined by measuring the gaseous daughter of radium, radon-222, in the exhaled air [34, 35].

d. Biopsy Samples

Biopsy samples are primarily of interest when the radioactive material is introduced in wounds. Where feasible, the wound area should be surveyed in order to determine the quantity of radioactive material present. Survey techniques which provide the required sensitivity for in vivo measurements are available for nuclides which emit gamma radiation or low energy photons [36]. In the event that the wound is found to be contaminated, surgical excision of the area may be required. In such a case, the nature and magnitude of radiation hazards are such that relatively radical surgery will seldom be indicated, and a decision should be reached only after consultation with physicians whose experiences in surgery and in radiobiology permit adequate balancing of the relative risks of excision and leaving the contaminated material in place. Following any such excision deemed necessary, the wound area should be resurveyed and the tissue removed should be analyzed.

e. In Vivo Measurements

Procedures are now available for sensitive measurements of gamma emitters and many beta emitters by the measurement of the radiation emitted from the body [37, 38]. This procedure has been used on a limited basis for the measurement of iodine-131 in the thyroid for many years [39]. In such tests, however, it is important to determine that the sensitivity of the measuring equipment is adequate to measure the small quantities which are permissible in the body [17]. The devices presently available for measuring the radiation from the body are, in general, elaborate and require massive shields to control the background to a steady and low level. The counters are massive liquid or plastic scintillation devices or large crystals of sodium iodide. The large sodium iodide counter, in addition to providing an estimate of the quantity of radionuclide in the body, also provides a spectral analysis of the radiation from which the radionuclide can be identified [37]. Such devices are expensive and require skilled personnel to operate. Many AEC and other installations now have such facilities. In the event of serious accidents, arrangements may be made to have individuals checked by one of these devices. Further discussion of such measurement techniques is included in Handbook 80 [40].

4. Physical Safeguards

The design of the laboratory or facility should be commensurate with the hazards of the material to be handled. There are two main problems that require control:

1. To prevent overexposure from external radiation, and
2. To minimize the entry of radionuclides into the body.

It should be noted that work in a very well designed and equipped laboratory can still result in hazardous conditions during handling, if improper procedures are used. Conversely, a laboratory which is not designed for the purpose can be used to handle reasonable amounts of radioactive material safely if care is exercised. Further discussions of problems of laboratory design and handling of radionuclides are given in Reference [12].

Physical safeguards to control exposure to external radiation make use of the decrease in radiation intensity with distance from the source and with shielding around the source. Control of internal deposition depends primarily on limiting dispersal of the radioactive material into the air so that it can be breathed as well as limiting direct contact with the material. The first line of control includes physical enclosures to confine the radioactive material and proper ventilation patterns throughout the laboratory. Protective clothing serves as an additional safeguard when containment is inadequate and is insurance in case of accidental escape of radioactive material. Control measures are facilitated by proper laboratory design and availability of equipment.

4.1 Shielding

Shielding serves primarily to minimize the external radiation reaching the body. It may, however, serve also as a barrier to the flow of contaminated material into the work area. Table 4 illustrates some of the typical materials used for shielding of various radiations. The particular material to be used depends upon the complexity of the operations, the type of source, the size of the source, the size of the equipment, and the location.

In design of shields the shielding material should, if possible, be placed near the source. If adjacent rooms or spaces are occupied, care should be taken to protect all such areas. Often the simplest method is to surround the source with the shield. Thus, if intense sources are handled in laboratory hoods with built in shields, it may be necessary to provide a shielding roof to protect individuals on the floor above. In many cases a direct shadow shield is not adequate due to the scatter of penetrating radiation from objects not in the beam between the source and the user. For large sources where entrance through the shield is required, a labyrinth should be provided to reduce scattered radiation. In construction of shields, particularly from objects such as lead bricks, care should be taken to see that cracks do not extend through the shield. If a radiation beam is necessarily brought into the room, the location of the beam shall be marked so that individuals do not inadvertently walk into it; and the beam should be stopped by a shield behind the point where it is used.

TABLE 4.—*Typical shielding materials for radionuclides*

Radiation	Shielding Material[a]		
	Permanent	Temporary	Additional clothing
Alpha	Unnecessary	Unnecessary	Unnecessary
Beta	Lead,[b] copper, iron aluminum, concrete, wood.	Iron, aluminum, plastics, wood, glass, water.	Leather, rubber, plastic, cloth
Gamma, x-rays	Lead,[c] iron, copper, lead glass, heavy aggregate concrete, aluminum, ordinary concrete, plate glass, wood, water, paraffin.	Lead, iron, lead glass, aluminum, concrete blocks, wood, water.	Lead fabrics (but not for "hard" gamma, see subchapter 4.3)

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[a] Arrangement of materials in general order of increased thickness required.

[b] Care must be taken with high atomic number materials to see that the bremstrahlung (x-rays) generated do not add significantly to the resulting dose after the beta rays are absorbed.

[c] For close work, lead is frequently backed with another shield, such as 1/8 in. iron or aluminum, to absorb secondary photo-electrons.

Although shielding can be calculated by an expert with a reasonable degree of accuracy, the transmission through a finished shield during conditions of use should be measured. If necessary, further shielding should be used or further restrictions imposed. Usually, however, time-consuming computations of the shield thickness for small laboratory experiments are unnecessary. It is frequently sufficient to estimate the required shielding and then add to it if measurements show it is inadequate.

In general, conservative calculations with generous provisions of shielding will take care of scattered and secondary radiation. Where permanent installations of maximum economy are planned, other references [41, 42] including NBS Handbooks 50 [43], 73 [6], 55 [44], 76 [8], and 63 [10] should be consulted.

Separate shielding considerations are involved for each of the radiations which are encountered in the laboratory.

a. Alpha Radiation

Usually no shielding is required for pure alpha rays because of their extremely limited range. However, containment to limit contact with the materials may be required to keep alpha emitters from being taken into the body.

b. Beta Radiation

The penetrating power of beta radiation depends on the energy of the beta particle. At average energies, a few millimeters of shielding will stop the radiation completely; and even for the maximum energies encountered with radionuclides, a few centimeters will suffice. The range of electrons in aluminum is given to within five percent by [45]:

$$R=412\mathrm{E}^{1.265-0.0954\ \ln \mathrm{E}} \qquad 0.01<\mathrm{E}<2.5$$

$$R=530\mathrm{E}-106 \qquad E>2.5$$

In the above equations R is the range expressed in mg/cm^2 and E is the maximum energy in Mev. The range in other materials is roughly the same when expressed as mg/cm^2. Figure 1 indicates the thicknesses of various materials required to completely stop beta radiation. Since beta particles are absorbed approximately exponentially over the first portion of the range, large reductions in the dose rate will be made even with thicknesses of material less than those given in figure 1. Some protection is afforded the body by normal clothing and, in cases where pure beta emitters are being handled, the use of heavy gloves such as leather or heavy rubber gloves will significantly reduce the dosage rate to the hands. Since the beta radiation affects primarily the skin, the maximum permissible levels for external beta radiation are greater than for gamma radiation for all regions of

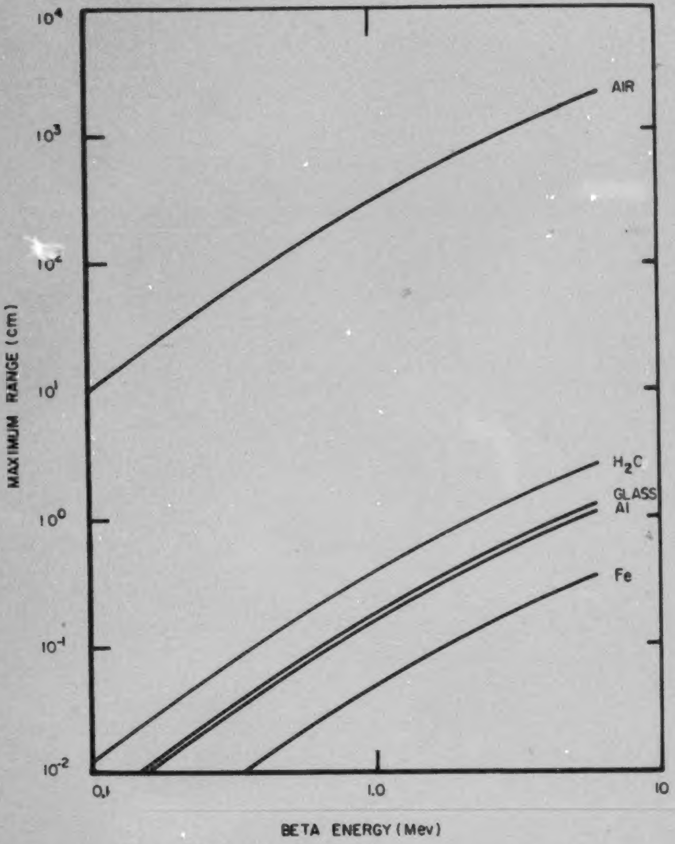

FIGURE 1. *Penetration ability of beta radiation.*

(Curves for aluminum and air based on Etherington, Ed. Nuclear Engineering Handbook, McGraw-Hill Book Company, Inc., New York, 1958). Other curves computed from that for aluminum on the basis of densities.

the body with the exception of the eyes (see section 2). For this reason, safety glasses, optical glasses, or goggles should always be required for work with beta emitters. In using leaded rubber gloves for handling beta emitters, do not overlook the possible presence of a gamma component of high energy. Gloves of this nature will serve as a scattering screen for the gamma photons, possibly resulting in a net increase in dose to the hands from the secondary radiations.

c. Gamma Radiation

The primary gamma radiations are absorbed in a shield in an exponential manner. This absorption can be characterized (for a narrow beam of radiation) by an absorption coefficient in the equation:

$$I/I_0 = e^{-\mu x}$$

where I is the exposure rate after passing through the shield, I_0 is the initial exposure rate at the shield, x is the thickness of the shield, and μ is the absorption coefficient.

The absorption coefficient can be expressed in units of reciprocal length or of reciprocal mass per unit area. Tables of absorption coefficients for several common shielding materials are given in appendix A [46].

In designing a shield, complications arise from the scattered radiation. In most cases, the actual radiation passing through the shield is higher than is indicated by the exponential absorption equation. This is due to scattered radiation reaching the observer from other portions of the shield. Such scattered radiation is taken into account by inclusion of a "build-up factor" in the equation. This is the factor by which the radiation calculated by direct exponential absorption of the primary beam must be multiplied to give the quantity actually penetrating, including the scattered radiation. Build-up factors for several energies of radiation are given in appendix B [47].

Calculations are available for specific geometries of source and shield. The figures in appendix C are for point gamma sources and a slab shield. These nomograms may be used to estimate the thickness of shield required to reduce the dosage rate of a given energy of radiation by the factor desired.

4.2 Containment and Ventilation

For low-level work, involving a few microcuries of most materials (table 2), an ordinary fume hood such as is used in chemical laboratories may be used to contain the radioactive materials and prevent contamination of the worker and the laboratory. If the quantity of radioactive material being handled increases, more elaborate containment measures, along with shielding, are needed. In much of the work involving the measurement of relatively small quantities of radioactive materials, more positive containment provisions are required to prevent cross-contamination of samples than are required to protect personnel.

In general, the ventilation system should be designed to permit air flow in such a direction that any radioactive material picked up by the air will flow away from the worker. In the design of new installations, the air flow should always be from a noncontaminated area toward the contaminated or potentially contaminated area. Poor ventilation is difficult to correct after an installation has been completed. It is always more satisfactory to install a correct system initially, than to try to overhaul a poorly designed setup. A good system of laboratory ventilation will confine the toxic contaminant, exhaust it with suitable duct work and fans, and pass this material through a collector or scrubber as needed before releasing it to the neighborhood. The design of a proper ventilation system will also provide sufficient air to make up for the amount exhausted.

Several excellent articles and texts have appeared on the subject of laboratory ventilation. References are given for those who are interested in more details on the design of ventilating systems for handling radioactive materials [48,49,50,51].

a. Hood Design

A laboratory hood is a simple enclosure in which work can be carried out without toxic materials escaping. Materials in the enclosure can become air-borne and escape by agitation from chemical or mechanical action, by thermal action from chemical reactions or heating devices, and by the syphon action from cross-currents of air. In order to keep the material from escaping from the enclosure, sufficient air should be exhausted to create an indraft through the face of the hood. This indraft must be strong enough to overcome the actions which tend to allow materials to escape. For handling low to moderate levels of radioactive materials, the average velocity through openings in the hood must be 100 fpm. For highly toxic or high-level radioactive material, the velocity through openings must be raised to an average of 125 to 200 fpm. At excessively high velocities, on the other hand, radioactive materials can be drawn out of open containers, contaminating the entire hood area.

In some cases, faulty circulation of air through the hood opening can be improved by increasing the volume exhausted and thus the velocity through the hood face, by restricting the opening or by providing streamlining baffles along the edges of the air opening. Instrument checks on the velocity of air entering the hood should be performed under the various conditions encountered during actual operations. Checks of air flow patterns with a small source of smoke can indicate the presence of cross-drafts and the possibility of pulling material from the hood.

The placement of the hood in the laboratory is important in respect to cross-drafts, which can pull material out of the hood. In general, a hood should be located well away from the doorway where the supply air must enter.

In some cases, during periods when the hood is unattended, it may be practical to use somewhat lower velocities, 75 to 80 fpm. Dual speed fans will permit operation at the higher velocity while the hood is in use and at the lower velocity when it is closed.

b. Glove Boxes

Elements emitting only alpha particles (or very soft beta rays, i.e., H^3 or C^{14}) can be handled even at high levels in completely enclosed containers known as glove boxes. Rubber gloves extend through hermetically sealed ports into the box and enable one to handle the radioactive material without contaminating his hands or lungs. For harder beta rays and gamma rays, gloves do not offer sufficient protection, and they are frequently replaced by mechanical manipulators operated from outside the box. This complication, along with the shielding required for medium or high-level work, makes the dry box much more expensive than the larger chemical hood.

Glove boxes should be provided with air locks through which samples can be inserted or removed. In such air locks, the sample is inserted in the air lock through one door which is then closed and the other door opened to remove the sample. Since all facilities are totally enclosed in the hood, it may be desirable to provide exterior controls for all services such as water, gas, electricity, etc.

Glove boxes are frequently provided with exhaust ports or fans and filters. Exhaust volumes of 20 to 30 cubic feet per minute will maintain a 50 feet per minute velocity indraft at any opening in a typical size and design of glove box. If intake filters are used, they should be located in such a position that the worker's body will not be exposed to the escaping material if there is an explosion or surge of pressure which could rupture the filter.

c. Exhaust Systems

The exhaust system is designed to remove from the laboratory the air-borne materials which are picked up in the hood. To safely vent the contaminated air, it may need to be filtered, scrubbed (or otherwise treated) and discharged at such velocities and elevations that it will not reach ground level at more than maximum permissible concentrations. Cleaning equipment must be selected with a view to the corrosive and toxic materials handled and the varying requirements for removal of radioactive materials. A choice of improper cleaning equipment will frequently result in efficiencies lower than needed or rapid deterioration of the cleaning material. Filters are available to achieve the high decontamination necessary for radioactive materials and for the particle size range which results in the greatest retention following inhalation exposure.

In a proper laboratory ventilation system, the duct work inside the building is under negative pressure. Under these conditions, any leakage due to poor construction or corrosion of the duct system will be into the ducts and the radionuclides will be confined. To accomplish this, the fan must be located not on top of the hood but outside the building or at the point where the exhaust leaves the building. Although this may require weatherproofing the motor, it can be an advantage when flammable material is handled because explosion-proof construction is not then required for the motor. Duct work connecting several hoods should have streamlined connections. Branch ducts should enter at angles of 30 to 45 degrees in order to permit better passage of air at high velocities. In such multiple installations, care should be taken to see that the exhaust system is balanced so that one hood does not provide the bulk of the air for the system.

Velocities of air in ducts must be great enough to maintain minimum transport velocities for the material being conveyed. Usual range of transport velocities for particulate material is 3500 to 4500 fpm.

In hoods where large quantities of water are handled, it is necessary to provide some means of removing the condensation which collects in the duct. When the system is intended to handle corrosive materials, the duct work should be of material resistant to corrosion.

The discharge should be at least five to ten feet above the laboratory roof, located so that fumes will not be carried back into the laboratory or into the air intake of adjacent buildings. Caps for weather protection obstruct the exhaust, directing it back down to the roof where it may be carried into the air intakes. The most practical fan discharge is the straight vertical stack with no directional baffles or projections. In this case, there must be a suitable drain connection in the housing. When necessary to discharge high level wastes to the atmosphere, independent stacks may be located downwind from the tallest nearby building at a distance of two or three times the building height. Details of the meteorology and dilution of radioactive materials in the atmosphere are found in other references [53,54].

Clean air must be supplied to replace the air in the room with an exhaust system, if contamination control is to be successful. If adequate air is not supplied to the room, the capacity of the exhaust system and the air velocity at the face of the hood is reduced. If there are multiple exhaust hoods and no makeup air, the airflow may be reversed through a hood that has a smaller fan or is turned off.

In building design, ventilating air may be admitted to offices, corridors, etc., and exhausted through rooms for low and high-level work in order, by control of static pressures. In a large laboratory with several rooms for radioactivity work, controls must be supplied to properly balance the flow of air from one room to the next and from one hood to another. High velocity air movements can be responsible for spread of contamination and should be avoided.

4.3 Protective Clothing

Special clothing which can be easily laundered or disposed of shall be worn when there is a possibility of contamination with hazardous amounts of radionuclides. The degree of protection required is a function of the quantity of radionuclides, the types of radionuclides, the nature of the operations being carried out, and the design of the laboratory and facilities available (see section 1.5).

Special clothing is not required for handling radioactive nuclides in sealed containers, except when they are opened purposely or accidentally.

In general, the aim of proper laboratory design and work procedures is to keep the occupied areas clean so that protective clothing is only required in the event of an accidential release of radioactive material. Frequently, however, the spread of radionuclides out of the laboratory is prevented by the use of special clothing which is worn only in the laboratory and not in the other portions of the building.

a. Garments

In general, protective clothing is not required if the quantity of radionuclides being manipulated amounts to less than the maximum permissible body burden based on the most critical organ as given in Handbook 69, [17].

For low-level work as characterized by table 2, it is advisable to wear laboratory coats or coveralls. Shoe protection, in the form of simple cloth or plastic bags to slip over the shoes, should be available in case the floor becomes contaminated. Rubber or plastic gloves are advisable if the radionuclide is being manipulated in chemical solutions or in dry or powder form. The wearing of such gloves will frequently save considerable time in decontamination of the skin and will assist in preventing the spread of radioactive materials to other parts of the laboratory.

For medium level work, coveralls, caps, gloves, and either special shoes or shoe covers are recommended. Again, if the laboratory design is such that the material is completely confined, laboratory coats may be adequate and caps and shoe protection need not be used except when the material escapes from the confinement.

If a person actually enters an area where there are highlevel quantities of uncontained radionuclides, he may require several layers of protective clothing including caps, shoe protection, several pairs of coveralls, several pairs of gloves, and respiratory protection. This type of work should not be encountered in normal operation but usually occurs during maintenance or clean-up procedures in facilities using large quantities of radionuclides.

Protective clothing also serves the function mentioned before of preventing the cross-contamination of samples. Protective clothing is often worn in low-level laboratories for the purpose of keeping individuals from bringing radioactive materials into the laboratory rather than for personnel protection.

Special disposal and laundry facilities are frequently necessary to handle protective clothing worn in work with radioactive materials. Handbook 48 [55] discusses such details as the monitoring of clothing before or after laundering, provision of change rooms or stations, showers, used clothing hampers, etc., and suggests laundering procedures which may be satisfactory or can be modified for a particular contaminant or installation. Commercial laundry service for contaminated clothing is available in many areas [3].

b. Respiratory Protection

Respiratory protection shall be worn when the concentrations of radioactive materials in the atmosphere, averaged over the working time, exceed the limits given in Handbook 69 [17]. Respiratory protection can vary from simple respirators which simply filter particles out of the air to completely self-contained or supplied air masks. Respirators should be in a form approved by a recognized laboratory for the type of service involved [56]. The limitations of the respiratory equipment must be known before use. Respirators and masks should be individually fitted and tested for rightness of fit by attempting to breathe with the inlet closed off before use. Reliance on masks in highly radioactive environment should be tempered by realization that even well-fitted masks can have leakage rates of one to two percent. The responsibility for seeing that the mask or respiratory protection fits shall be with the individual worker. It may be noted that growth of beard may prevent a tight fit of the mask. In general, masks with a separate source of air supply are used only in high-level work where the individual actually enters a highly contaminated region for maintenance or clean-up work.

c. Shielding Garments

For close or contact work with radioactive materials emitting radiation of low penetrating power, shielded clothing such as leather or leaded gloves and aprons or eye protection may be used to increase allowable exposure time. Leather and rubber are effective against most beta radiations. Fabrics loaded with high atomic material, such as lead, are used for shielding against scattered x-radiation in fluoroscopy. At higher energies, the great increase in weight and loss in flexibility which would be necessary to shield against gamma-rays rule out the use of shielding garments.

4.4 Laboratory Design

In work areas where unconfined radioactive material is handled, surfaces of floors, benches, hoods, and other work areas should be impervious or easily replaceable for economical control of radioactive contamination. National Bureau of Standards Handbook 48 [55] discusses recommended and unsuitable surfaces. For example, uncoated wood, concrete, soapstone, and plain rolled metals absorb radioactive materials. For the same reason, however, absorbent paper with a waterproof backing is very useful as an easily replaceable temporary covering. Absorbent paper should be replaced at frequent intervals and should not be allowed to accumulate radioactive materials until it becomes a dust hazard. Tile is recommended for work surfaces because it is easily replaced in small sections. Subject to special requirements, a general order of preference for tile material is: vinyl plastic, asphalt, rubber, ceramic. Linoleum gives acceptable service if properly cared for. Polished stainless steel, plate glass, tempered glass, and die stock masonite each have advantages. Porcelain has limited uses.

Ordinary paints, varnishes, and lacquers are not recommended for wear surfaces. Tygon paint is suitable for some work surfaces, and for coating instruments and equipment. Strippable plastics are used extensively but have low wear and tear resistance, and are soluble with some organic solvents. Handbook 48 [55] suggests some suitable commercial paints and coatings, and more are continually being added to the market.

Dust-collecting surfaces should be eliminated as much as possible. Lights should be recessed, pipes enclosed, and shelves covered by doors. Cove corners between wall and floor facilitate housekeeping and decontamination. Cracks should be filled and sealed. Ceilings are always advisable, to prevent contamination from settling on roof trusses.

Experience in handling unconfined radioactive materials has shown that where sizeable quantities are used, special facilities for disposal of liquid radioactive wastes from housekeeping and decontamination operations should be provided, both for radiation protection and economy of operation and maintenance. Where only small amounts of radionuclides are used as auxiliary tools for research, medical treatment, or process control, such special facilities may not be required if the necessary protection and control can be achieved by special instructions and procedures. (See Handbook 49 [57] and 53 [58] on disposal of phosphorus-32, iodine-131, and cesium-137.)

Separate sewer systems, for rooms and facilities where liquid radioactive material may leak, spill, or be disposed of over a period of years, permit better radiation control and enable provision of more economical waste treatment facilities if necessary. Sinks, wash basins, and floor drains to be used for radioactive waste should be plainly marked. Wash basins, where contaminated hands may be washed, should have knee- or foot-operated faucets. Such pipelines and sewer systems may need special design considerations. Because of deposition of the radioactive material on the pipe wall, the radiation from the pipeline itself may become excessive. It may be economical to reduce maintenance by using piping which can be easily decontaminated, such as stainless steel.

Good permanent records of pipeline locations are important, not only to facilitate maintenance but also to prevent damage to the pipeline during repairs in the vicinity. Code painting of such pipes to permit identification may be advisable [59]. Further details on waste disposal are given in section 8.

4.5 Laboratory Facilities

The safe handling of radionuclides requires the use of special equipment appropriate to the materials used, their level of activity, and the type of operation. For laboratory operations, tongs, forceps, trays, and mechanical holders are needed. Semiremote-control sampling, stirring, and pipetting devices are included [60]. Mouth pipetting shall be strictly forbidden.

Low-level amounts of soft beta emitters, i. e. 100 millicuries or less of hydrogen-3 and 10 millicuries or less of carbon-14 and sulfur-35 may be used with minimum facilities and moderate precautions.

Long-handled tools provide reduction in dose rate by distance for handling millicurie amounts of beta or gamma emitters. Operations with multicurie quantities behind shields require use of specially designed remote-control tools and shielded optical systems, such as lead-glass windows, periscopes, or mirror arrangements.

Magnetic, electromagnetic, and pneumatic principles are applied to the remote handling of radionuclides.

Shielding casks or containers for the movement or storage of radioactive materials are of many shapes and sizes but are characteristically compact to minimize weight. Inner containers for liquids should be nonbreakable, either metal or polyethylene, for example, or else completely surrounded by absorbent material sufficient to absorb the entire liquid contents and of such a nature that is efficiency will not be impaired by chemical reaction with the contents.

Capsules for radioactive sources require special design considerations to control decomposition, as explained in Handbook 73. [6]

4.6 Isolation of Facilities

All areas in which unsealed sources of radionuclides are handled shall be conspicuously posted in order to warn against possible radiation hazards and the possibility of encountering radioactive contamination. Such marking, specifying the low level of hazard, might be advisable, even when the quantities being handled are so small as to present no hazard, in order to warn against unauthorized handling or movement of samples or equipment with consequent possible spread of even small amounts of radionuclides to other areas.

Areas in which the radiation dosage or the contamination levels are significant shall be marked and also provided with a physical barricade to prevent inadvertent entry. The type of barricade should be commensurate with the hazard considering the quantity and accessibility of the radioactive materials and the radiation levels. Temporary barricades can include posts and rope, chain or colored tapes. More secure barricades could include permanent fences, separate rooms, or vaults. Whenever the radiation levels are high enough to produce serious exposure in a short time, the area shall be locked to prevent inadvertent entry.

Areas in which the radiation levels or contamination levels are sufficiently high to permit an individual to receive a significant portion of the maximum permissible dose during normal working hours should be designated as restricted areas with entry controlled by prearranged procedure. Such controlled areas have been designated as "radiation zones," "contamination zones," etc., with the criteria as to dosage rate and contamination level selected according to local conditions. In many instances, the restricted areas are defined at such levels that the radiation field outside of these areas is minimal. A typical criterion for a restricted area might be: "an area where the radiation level can exceed one mrem per hour, or where there may be significant surface contamination or where air concentrations can exceed one-tenth of the applicable maximum permissible concentration." The intent is to provide a defined region where precautions are necessary for safety. Areas in which the defined doses can be exceeded due to changes in conditions should also be controlled. "Significant surface contamination" is not precisely defined, but depends upon the radionuclide involved, the radiations emitted, the degree of fixation, ventilation, and accessibility. It is impractical to list general numerical limits which might not be absurdly restrictive, in many cases, or to list specific limits for all possible cases. Typical limits as well as further discussion of the problems are given in NBS Handbook 48 [55] and in section 5.6 of this handbook.

The restricted area should be designated by the radiation symbol approved by the ASA in ASA N2.1-1960 on signs, tags, stickers, etc., posted in sufficient quantity so that at least one symbol is easily visible from any ordinary angle of approach. The wording should explicitly warn of radiation hazards. Unexplained wording, such as DANGER, is not appropriate. (See also Handbook 61 [61] on radiation information labeling).

4.7 Storage

Radionuclides, when not in use, shall be stored in such a manner as to preclude the possibility of inadvertent radiation exposure of individuals either from external radiations or from movement of the radionuclide from its container into the laboratory air or onto surfaces. It is advisable to provide designated storage locations which are adequately shielded and, if necessary, ventilated. Separate ventilation is required for the storage area where the material could escape from the storage vessels; when the storage area is used for vessels contaminated on the outside; or when sealed sources liable to rupture and dispersion of their contents are stored. The inclusion of a floor drain flowing into an appropriate disposal system will aid decontamination proceedings in the event of a spill. These areas should be plainly marked with a radiation sign, and if access to the area is available to individuals other than those working with the radionuclides, a lock should be provided to prevent deliberate or inadvertent entry. Fireproof and waterproof storage should be provided for larger quantities of radioactive materials.

In many laboratories, such storage areas are provided by wells or holes in a concrete floor or wall with closure by step plugs to eliminate direct beams. For laboratories handling multicurie amounts of gamma emitters, a remotely controlled manipulator to remove the radionuclides from storage and place them in a shielded location is desirable. For less than curie amounts, the handling may be done with tools having handles long enough to keep the radiation dose received to an acceptable level. The use of long-handled tongs for handling glass vials should be minimized because of the possibility of dropping the vial.

For the storage of smaller amounts, a lead or iron safe or a specially designed shielded area may be used. The design of such devices should be such that adequate space is available for storing the radionuclides required and the space should be partitioned so that each sample can be easily located and records kept on the quantities available. In cases where a number of different samples are stored in the same area, shielding partitions or containers may be desirable to minimize total radiation received when removing one sample.

5. Procedural Safeguards

5.1 Operational Controls

Each controlled area (or group of controlled areas) shall be identified with appropriate signs. Such signs shall be in accordance with applicable requirements of any regulatory agency. All employees and visitors who enter a controlled area shall be informed of the pertinent requirements and procedures for the protection of themselves and fellow workers against internal and external exposure. In large installations a simple introductory handbill, pamphlet or lecture could be given to each new employee or visitor to inform him of any pertinent rules and regulations.

General rules and instructions should be written in detail commensurate with the hazards of the radionuclides used and the nature of the work and shall be made available to all employees. A current listing of the supervisor or other persons assigned responsibility for radiation protection and persons to be contacted in various emergencies shall be maintained. Floor plans and drawings of all facilities handling dangerous amounts of radioactive materials should be filed outside the facility so that this information will be readily available in the event of a fire or an explosion which makes an immediate on-the-spot study of the area impossible.

For repetitive work with radionuclides, the standard operating procedures should be established, preferably in writing. They may be incorporated with work procedures and other safety requirements. Written instructions not only aid understanding and compliance, but also indicate requirements for revision as the basic elements of the work change or as unforeseen hazards become evident.

When non-repetitive work or special jobs having unusual hazard potential are to be performed, a careful job hazard-analysis should be made by responsible individuals and pertinent operating procedures established, preferably in writing. Each worker should understand the requirements and should be expected to comply during the work. The resultant instructions should clearly define and limit the work to be done; specify necessary personnel monitoring equipment, protective clothing and respiratory protection; state the maximum average radiation survey readings and the exposure time limits. If the readings and limits are unknown in advance, the work should not be started until a survey is made; and, if necessary, continuous or intermittent monitoring should be performed. Personnel should be continually informed of such readings and limits during the work so that overexposure is prevented. Brief instructions in case of emergencies should be included. Finally, the instructions should specify any check-out requirements for leaving the area such as personal contamination surveys, removal of contaminated clothing, etc. A convenient form for providing such instructions for non-routine work is given in appendix D.

Other forms which might be adopted for radiation control include radioactive shipment records for transfers of material, inventories of radiation sources, records of waste disposal, and records of significant exposure of visitors.

When cumulative exposures are expected to be close to the permissible limits, a daily estimated exposure record should be maintained for each person. This will permit better control of his exposure and aid in interpretation of personnel meter results. During complex work in the presence of one or more high activity sources, a running exposure record is suggested. Separate records of exposures to limited portions of the body (with higher exposure limits) as well as to the whole body may permit more economical use of manpower.

The use of self-reading pocket dosimeters can supplant the running records sometimes required in complex work. In addition to their use on the torso, they can be fastened to hands, arms, legs, head, etc., to measure localized exposures.

5.2 Exposure Records

Permanent records of exposure to radiation and radioactive material should be maintained for each exposed person as determined by personnel meters and bioassay. These are supplemented by radiation survey and estimated exposure records obtained during the work. Typical forms which can be used for this purpose are given in appendix D.

Regular examination of the exposure records warns of impending overexposures and of jobs or locations where control can be improved. Compilations from the records can yield more economical design of new or revised facilities, and permit scheduling manpower to allow for expected exposures. An individual's record is valuable for medical and legal purposes. Records should be obtained of significant exposure received by employees in previous occupations, in visits to other sites, or in service in the armed forces.

5.3 Investigations

Overexposures, serious accidents, and spills of radioactive materials should be investigated impartially to permit correction of any conditions which could have led to the event. Such investigations are particularly useful when they indicate defects in previously accepted procedures or equipment so that corrective action may be taken immediately. Information on the cause of the event should be disseminated so that others may profit by the experience.

The data obtained on such an investigation should be written in detail and made a part of the exposure record of each individual involved. Such information is frequently useful in reviewing the capabilities of individuals for handling radioactive materials, for assessing their total exposure history, and for interpreting results from later measurements of excreta.

Similar investigations should be made on those occasions when there has been a failure to use a personnel meter or when the record from an exposure meter is lost due to failure of the device or in processing. Although such an investigation cannot replace the lost record, it can frequently put an upper limit on the possible exposure and indicate whether an unusual exposure was probable. The best estimate of the exposure received should be entered in the exposure record.

5.4 Personal Decontamination

Special techniques have been evolved for the removal of radioactive contamination from the skin, hair, etc. The objectives of these techniques are to reduce radiation exposure promptly, to minimize absorption of radionuclides into the body, and to keep localized contamination from spreading. In each case, decontamination should continue until no activity is detectable with appropriate survey instruments (see section 6.0 and Handbook 48 [55]) but in no case shall decontamination continue to the point where the effectiveness of the skin as a barrier is destroyed. Open wounds through which the contaminant might enter the body must be protected.

Two points are of extreme importance in personnel decontamination:

1. A physician should be called as soon as it is seen that the few simple decontamination efforts which are safe for unskilled use are not appreciably reducing the contamination any further.
2. Decontamination efforts should cease when the skin starts to become thin and reddened. The health of the skin must be maintained to minimize absorption and internal deposition. If necessary, the contaminated area may be carefully covered by bandages so that the radioactive materials are not spread and the patient released until the skin is replaced. At this time, decontamination efforts may be resumed.

A record of the decontamination procedures and results is an important addition to the personnel exposure record. A sample form for this record is given in appendix D.

The following procedures have been effective in minimizing exposure and entry of radioactive material into the body during decontamination.

A. General

1. Make a quick survey of exposed skin, hair, and clothing with appropriate radiation measuring instruments to indicate where decontamination is most urgently needed.

2. Carefully remove contaminated clothing.

B. If Survey Shows Widespread Contamination

1. Shower with soap and water. Keep radioactive materials out of eyes, nose, and mouth and minimize spread to any clean area of the body.

2. Dry thoroughly.

3. Repeat survey.

4. If contamination is still widespread over the body, repeat the shower with scrubbing, taking care not to damage the skin.

5. Repeat survey.

6. If contamination still exists, select the most highly contaminated areas and start decontaminating as given below for localized areas.

C. If Contamination is in Localized Areas

1. Both the person doing the decontamination and the contaminated person should don suitable protective clothing such as laboratory coats, surgeons' rubber gloves, and respiratory protection, if needed.

2. Localize the area to be decontaminated so that the radioactive materials are not spread to other parts of the body. In many cases, the protective clothing will suffice for this purpose; in others, it may be desirable to use supplemental protection.

3. Decontaminate hair by repeated application of liquid soap and rinse water, using towels to keep water from running onto the face and shoulders. Acid goggles can be used to protect the contaminated person's eyes. Thoroughly dry the hair before resurvey. If the contamination is still present in significant quantities and is no longer being reduced after three such washings, the physician should be notified.

4. If contamination is found in the eyes, mouth, or an open wound, flush copiously with water and contact a physician immediately for further instructions.

If the contamination is in the nose, have the patient clean nostrils with wet cotton swab sticks, blowing his nose frequently and taking care to keep the radioactive materials out of his mouth. Deeper contamination may be removed by nasal irrigation which should be started as soon as possible after the suspected inhalation to minimize swallowing of contamination. This irrigation can be performed by inserting ¾ in. to 1 in. of thin gum rubber tubing (⅛ in. ID × 3/64 in. wall) into one nostril and permitting a normal saline solution (container positioned one to two feet above the person's head) to flow and be discharged from the other nostril. Whenever possible, all solutions from the nose or mouth should be saved for later radiochemical analysis. If any water goes over into the patient's throat and mouth, have him spit it out and flush his mouth without swallowing any water. In each of these cases, urine and fecal analyses or measurement of radiation from the body is desirable.

5. Possible contamination under the fingernails should receive particular attention. Cleaning and trimming the nails may be followed, if necessary, by the treatment described in D.

D. Removal of Localized Skin Contamination

If the contaminant is in a chemical solution, which might react with the skin, immediately dilute it with water, using dampened swabs to minimize spread of the contaminant. Maintaining portable kits containing supplies such as those listed in table 5 will facilitate decontamination. The two chemical agents, titanium dioxide and potassium permanganate, have consistently proven

TABLE 5.—*Typical supplies for skin decontamination kit*

2	pr. surgical gloves
2	pkg. cotton-tipped applicators
12	tongue blades
8	10-ml beakers
1	soft scrub brush
1	magnifying glass
1	1-qt. ice cream carton
1	box facial tissue
1	4-oz. bottle potassium permanganate [a]
1	4-oz. bottle sodium bisulfite [b]
1	1-oz. jar titanium dioxide paste [c]
2	4-oz. bottles distilled water
1	4-oz. bottle liquid soap (baby)
Instructions	

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[a] Saturated solution (approximately four percent) made by dissolving 5 grams $KMnO_4$ in four-ounce bottle leaving excess crystals in bottle. Replace when excess crystals disappear (few months).

[b] Four percent solution, made fresh when needed by dissolving four and one-half grams $NaHSO_3$ crystals (may be in ready-to-mix package) in a four-ounce bottle of distilled water.

[c] Made by mixing precipitated TiO_2 (very thick slurry, never permitted to dry) with a small amount of lanolin.

superior for decontamination. Swabbing the titanium dioxide paste on and off removes contamination lodged under scaly surface of skin. The permanganate solution dissolves that absorbed in the epidermis, removing a minimum of protective skin. The bisulfite decolorizes the permanganate stain. Detergents, wetting agents, or cornmeal base soaps may be employed instead of bar or liquid soap. A two per cent solution of salicylic acid in ethyl alcohol provides effective keratolytic action to remove surface skin, but this should be used only under a physician's direction.

Starting with soap, then going in order to titanium dioxide, permanganate, and bisulfite, each agent should be applied and rinsed off three or four times (with survey after drying) before using the next.

5.5 Contamination Control

a. Prevention of Contamination

From the safety standpoint, the control of contamination imposes the most exacting requirements in the handling of radionuclides. It is much more effective and economical to control contamination at the source than to decontaminate areas and equipment on a larger scale. Advance preparations should be made to minimize leaks, spills, or other losses. Areas where radioactive contamination is present or likely should be isolated from casual entry. The full cycle of protective clothing procedures may be needed: issue, donning, proper use, removal at the barricade, collection and laundering. Incidents can become accidents through inadequate planning and care at each step.

Likewise, proper preparation should be made for the careful handling of radioactive material based on its physical state: solid, liquid, or gaseous; powdered, corrosive, absorptive, etc. Drip pans, splash guards, backed absorbent paper, strippable coatings, and similar inexpensive provisions greatly reduce the need for decontamination, surveys, and replacement of permanent facilities. Well-channeled ventilation is a major aid to prevention of contamination (see section 4.2). Good housekeeping is well repaid. In every operation with radioactive materials, frequent use of survey meters will help to minimize personnel exposures and prevent contamination.

b. Decontamination

When prevention fails, decontamination begins. The techniques of decontamination begin with washing with soap and water and continue through the use of detergents, wetting agents, solvents, chemical solutions, and, in the case of contaminated equipment or laboratory surfaces, physical removal such as stripping, scraping, grinding, and sandblasting. The use of bag type vacuum cleaners should be considered for certain decontamination jobs. This is particularly true where the contamination is particulate in nature and likely to be soluble in water or other cleaning liquids. National Bureau of Standards Handbook 48 [55] recommends some of the more suitable techniques and agents for the removal of radioactive contamination. Commercial decontamination agents are continually being improved and the list for specific contaminants expanded.

Caution is required in the use of some organic and chemical solvents. Certain ones not only are toxic in themselves, but may increase the exposure hazard by increasing the absorption of radionuclides through the skin. It is generally true also that the properties which make a good decontamination agent result in poor removal of radionuclides in waste disposal treatments. EDTA (ethylenediamine tetra-acetic acid and its salts) is one of the best decontamination agents, but its presence makes waste treatment much less effective.

5.6 Control of Contaminated Articles

Wherever radionuclides are used, clothing, tools, and equipment may become contaminated. These contaminated articles should be controlled to prevent spread of radioactive materials to clean areas or even to public areas outside of the installation. The problems involved depend on the magnitude of the program and the experience of the individual involved.

In general, these items fall into two classes, namely: (1) those items which remain in a controlled area and are reused, and (2) those items which are no longer needed and can be safely released, after decontamination, for use in other work.

a. Controlled Area

Contamination levels in a controlled area are usually subjected to indirect checks such as measurement of air concentrations, frequent measurements of personnel contamination, etc., so that hazards induced by the presence of the contaminated article become a part of the over-all program of control. For this reason, controls on the levels of contamination on articles used in the laboratory can be set by considerations of over-all hazard and the protection program, and specific limits are not needed.

Equipment to be transferred from one controlled area to another should be thoroughly surveyed before moving and, if the level is higher than the general level in the new area, decontamination should be carried out. Equipment which is shown to be contaminated, or which has inaccessible parts and has been in a controlled area, should be marked with the radiation symbol. If it is to be stored or used by another group, it should also bear a description of the kind and level of contamination and the date of the survey. If the external radiation level exceeds one mrem per hour or if the contaminant is such that an individual might accidentally receive from it more than one-tenth of the maximum permissible body burden [17], then additional safeguards should be applied. These might include an outer container or shield or storage in a locked enclosure.

b. Conditional Release from Control

A conditional release procedure may provide suitable control of articles such as heavy mobile equipment which do not leave the installation, or of areas or fixed equipment not entirely free of contamination but causing insignificant hazard. Requirements for conditional release should include the following:

a. The equipment is not contaminated to a level where it is a radiation hazard itself.

b. The intended use presents no radiation hazard to informed users.

c. Regulations for controlling the radiation or contamination are securely attached to the equipment in a prominent place.

d. Property inventory records are maintained for each item listing its "home" location, radiation status, person responsible for control, and date of latest inventory.

c. Unconditional Release from Control

If articles are to be released from the controlled area for use in uncontaminated areas, surface contamination must not exceed acceptable levels. The permissible contamination depends on such factors as the relative hazard of the radionuclides involved, including both the external radiation and the uptake in the body; the degree of fixation of the contaminant; the mobility of the article involved; the accessibility of the contamination in the normal use of the article; and the possible interference with sensitive radiation measurements.

Measurements of surface contamination are usually expressed in terms of the response of the instrument used rather than in absolute units because of the unknown depths of penetration of the material into the surface. Suggested levels of "significant contamination," below which an item can be released from the controlled area, are given in table 6.

A "wipe" or "smear" test is valuable for detecting the presence of loose contamination. In this test, the object is wiped with a piece of cloth, paper, or sticky tape; and the material measured on a sensitive device which will detect the radiation emitted (see section 5.7). The degree of fixation of contamination and the mobility of the article are particularly important.

Requirements for unconditional release should include the following:

a. All accessible surfaces are free of significant contamination, as determined by surveys with sensitive alpha, beta, or gamma monitoring instruments, appropriate to the nuclides that have been used.

b. It must be reasonable to presume that inaccessible surfaces are uncontaminated, on the basis of two premises: that, without being cleaned, accessible surfaces are free of significant contamination; and that no radioactive materials could have contaminated the inaccessible surfaces without having contaminated the outer surfaces as well.

c. The materials of which the item is made are such as would not be likely to occlude radioactive materials.

TABLE 6.—*Suggested levels of "significant contamination"*

Measuring instruments	Level for Nuclides in groups	
	1 and 2[a]	3 and 4[a]
Geiger counter (β, γ)[b]	100 cpm	100 cpm
Ionization chamber	0.1 mrad/hr	1 mrad/hr
Alpha counter	1 d/m/cm^2	10 d/m/cm^2

[a] Table 2.
[b] Flat plate area of two square in.

d. Wipe or smear tests indicate no detectable loose contamination and there is reasonable assurance that any fixed contamination will not become loose and subject to spread at some later date.

5.7 Survey Procedures

Radiation surveys are performed to indicate the general level of radiation in the working area and to provide information on changing levels of radiation or contamination as the work progresses. Where only a few radionuclides in low or medium level quantities are handled, monitoring of the work and general contamination levels may be carried out by the worker, although occasional checks of the area should be made by an individual not directly concerned with the work. Where many radionuclides are used at high levels of activity, it may be advantageous to employ one or more persons especially to perform such monitoring and to assist the workers by monitoring during the performance of the job.

An area survey includes the initial examination of a facility when radioactive material is first introduced, and subsequent routine or special surveys to insure radiation and contamination control. Handbooks 55 [44] and 76 [8] recommend surveys to be made initially and subsequently for installations of radiation-generating machines; similar procedures should be employed where radioactive materials are handled.

Routine surveys should be scheduled in order to detect inadequately shielded radiation sources, excessive surface and air-borne contamination, and waste not properly disposed of. The unlikely and remote locations should not be overlooked. Lockers, offices, roofs, and vehicles are examples of locations where radiation control is occasionally compromised by inadvertent movement of contamination and should be surveyed on a routine basis.

Surveys for surface contamination should include smear (wipe) tests to determine how much is loose material which may become air-borne or otherwise transferred to personnel. A smear test is made by wiping the contaminated surface with a clean cloth or paper, or pressing a sticky tape on it to detect and estimate the amount of loose contamination which might be rubbed off the surface in subsequent usage. The test can be semiquantitative if the areas, materials, and instruments are standardized (e.g., 100 sq in. of painted surface is wiped with one square inch of paper and measured with an end-mica-window counter). Adhesive tape is one of the better collecting materials. Hard-finished paper is better than soft. By removing the smear sample to a region of low background for measurement, even vessels containing highly active sources can be checked for contamination not detectable by direct survey. The wipe tests may be adjusted to the hazard of the contaminant and the usage of the surface.

Measurements of radiation levels and contamination should be made at intervals during the conduct of the work, particularly when conditions change drastically such as when source containers are opened or material is transferred from one vessel to another. Such measurements may be made by the worker, although for tasks involving his full attention with very active sources, it may be advisable to have another person assist by monitoring.

At the conclusion of the work or before leaving, for example, to eat or smoke, surveys of the clothing, skin, and hair should be made to prevent possible movement of radionuclides to other parts of the building and further exposure of the individual.

5.8 Air Monitoring

In regions where radioactive materials can become airborne, air samples should be taken with samplers designed to collect the contaminants involved. Filter type samplers, impactors, and electrostatic precipitators are used to collect particulate materials; vapors and gases require special techniques involving absorption or chemical reaction for collection [62, 63]. Nonmetabolizing gases (such as argon, krypton and xenon) ordinarily result in higher dosage rates from the external radiation to a body immersed in the gas than the dosage rates from internal deposition. Monitoring for these gases involves measurement of the whole-body exposure with no need to collect samples except in very small enclosures, where the external dosage rate is lower than the internal.

Air samples collected on filters or electrostatic precipitators will contain the decay products of the naturally occurring radon and thoron [64]. These daughter products are frequently in sufficient quantities to mask the contaminant being investigated. Since the longest-lived naturally occurring contaminant collected on the filter is ThB with a half-life of 10.6 hours, the long-lived activity can be obtained by measurement after the ThB has decayed or by taking two measurements spaced in time adequate for significant decay of the ThB and solving the two simultaneous equations for the long-lived contaminant.

$$D_a = \frac{D_2 - D_1 e^{-\lambda \Delta t}}{1 - e^{-\lambda \Delta t}}$$

D_a = disintegration rate of long-lived alpha emitter.
D_1 = total disintegration rate measured on first count.
D_2 = total disintegration rate measured on second count.
Δt = time interval between measurements.
λ = disintegration constant for ThB ($0.693/10.6\ \mathrm{hr}^{-1}$).

The first measurement should be made at least three hours after sampling stops and the second 10 to 24 hr later.

Since the maximum permissible levels for beta emitters are several orders of magnitude higher than for alpha emitters, daughters of radon or thoron are not as significant in measurements of beta emitters.

Such filtering procedures provide indications of levels in the air only at some time after the sample has been taken. The results must then be used to indicate areas where high concentrations are probable and where respiratory protection is needed. When starting new work where significant air concentrations could occur, respiratory protection should be used until air samples confirm the absence of a hazard. Recent instrument developments include impaction devices which discriminate against the radon daughters by taking advantage of the small size of the dust particles carrying them [65]. Such a device should be used as the sole monitor only when it has been shown that the alpha emitter being monitored is not present as very small particles.

The concentrations determined by air monitoring should be compared with the maximum permissible concentrations (MPC) given in Handbook 69 [17]. If these concentrations exceed the MPC, corrective action should be instituted and respiratory protection provided.

If the identity of the radionuclide involved is not known, it should be assumed that it is the most hazardous of those possible for purposes of estimating the MPC. Where large numbers of different nuclides are involved with no knowledge of the ones present in the air sample, a provisional limit of 10^{-9} μc/cc for beta emitters and 10^{-12} μc/cc for alpha emitters may be used. As information on specific nuclides becomes available, the provisional limit should be replaced by a limit calculated for the known mixture according to the principles given in Handbook 69 [17] and section 2.3 of this handbook. It should be noted that these provisional limits are provided solely to minimize the necessity for identifying radionuclides present in the air when concentrations are low and should not be used as a criterion of hazardous conditions when additional information on the composition is available. In many cases, it is possible to identify shorter-lived nuclides (such as the daughter products of thoron) by simple techniques such as decay curves. In this case, the provisional level should be applied only to the unidentified portion.

5.9 Emergencies

Even in a carefully planned and executed program, unforeseen events may lead to accidents involving the spread of contamination or excessive radiation levels. Such accidents may involve merely spills of material in the laboratory where prompt action will minimize the movement of radioactive materials or they may involve extensive damage to the facility from floods, fires, earthquakes, or explosions. In any emergency, primary concern must always be the protection of personnel. This will usually involve confinement of the contamination to the local area of the accident whenever possible.

Consideration of possible accidents and methods of handling them should be an important part of designing the laboratory rules and programs. Only general guides to actions in an emergency situation are given since the numerous differences in laboratories, personnel, and programs make the specification of detailed plans applicable to all situations extremely difficult. Plans shall be devised ahead of time for coping with probable accidents. All persons working with radioactive materials or involved in case of accident shall be informed as to action they should take to minimize exposure.

All accidents should be investigated as soon as possible and the complete report made a part of the laboratory records.

a. Preventive Measures

Many steps can be taken ahead of time to avoid accidents involving radioactive materials or to minimize the consequences, if they do occur. New techniques or procedures should be thoroughly tested with inert materials ahead of time and approved by the person responsible for radiation protection. In many cases, it will be advisable to rehearse the entire procedure before using radioactive materials. Steps in the procedure likely to give trouble or result in a spill can be identified and improved as a result of such rehearsals. Protective clothing suitable to the conditions expected if an accident occurs should be worn.

Features for preventing the spread of contamination following an accident can be built into a laboratory or facility. Containers for radionuclides should have double integrity either in the container or in absorbent material in the container so that breaking a single vessel will not release the contents. Emergency exits should have quick closures, and ventilation shutoffs should be readily accessible. Fire-proofing is advisable, particularly in ventilation ducts and filters. Rapid and secure shielding of radiation sources should be available, and the sanitary water and sewer systems should be isolated from possible contamination.

b. Emergency Procedures

When an accident such as a spill could result in overexposure or contamination of workers, procedures for dealing with the emergency should be planned and rehearsed. In the event of a spill, workers should leave as rapidly as possible, closing the doors behind them. They should not leave the immediate vicinity, however, until surveyed for contamination on the shoes, clothing, skin, and hair in order to prevent spread to other areas. Contaminated clothing should be removed and left behind and areas of contamination on the body cleaned immediately. Reentry into the area should be made cautiously with survey readings to define the area of contamination and radiation levels. If air contamination is present above the MPC, respiratory protection should be worn.

Arrangements shall be made with the local fire and police departments to handle emergencies such as fire or flood. Personnel of these departments should be briefed on the types of sources stored in the area, their location, possible special hazards, etc., and cooperative plans should be made. Protective clothing and personnel monitoring equipment should be available for these groups. There are several publications on radiation fundamentals for fire departments which could be used to provide training and details to the groups involved [66, 67, 68]. In addition the U.S. Atomic Energy Commission has several programs to assist fire departments in preparing to handle fires where radionuclides are located.

6. Radiation Instrumentation

The success of any radiation protection program depends ultimately upon the ability to measure the quantities of radiation and radioactive materials to which people are exposed. There is a wide variety of instruments available for measuring various types of radiation under many different conditions. In many cases, however, the interpretation of the readings from these instruments in terms of the actual radiation exposure of various portions of the body and the actions required in order to remain within necessary limits is not clear so that these instruments should be used by people trained in the interpretation of readings.

Developments in the field of instrumentation are rapid with many new devices becoming available for survey, detection or monitoring. In many situations the new plastics which are similar to tissue in their interactions with radiation permit the measurement of the radiation dose in rads [69]. Detailed discussions covering the basis of dosimetry as well as the application of instruments may be found in many articles [5, 9, 40, 70, 71, 72].

6.1 Personnel Meters

Personnel meters are devices to be worn on the person for the purpose of measuring the radiation received in the course of the work. For situations where generalized total body radiation is received, they are generally worn on the upper torso. However, where localized doses to the hands or other portions of the body are possible, additional devices should be provided in order to monitor exposure to these portions of the body. When working over a shield, a person will often find his exposure limited by the exposure of the eyes, and he should therefore wear a dosimeter on his forehead. Monitoring of personnel with ionization chambers or film badges is not necessary if soft beta emitters (e.g., hydrogen-3, carbon-14 and sulfur-35) are the *only* nuclides used. In fact, film badges and dosimeters, unless specifically designed for these energy levels, are unsuitable.

Handbooks 51 [5], 55 [44], 57 [73], 59 [7] and 76 [8] contain descriptions of personnel meters and recommendations for their use and evaluation of readings.

a. Ionization Chambers

Ionization chambers are used as personnel meters primarily in situations where the exposure is to gamma or x-ray doses. These chambers can be constructed so that an auxiliary meter can measure the loss of charge or a quartz fiber electroscope can be built into the ion chamber and the instrument may be read at any time without auxiliary equipment. The self-reading device is frequently used in situations where a close control of the radiation exposure at the time of work is desired. Ionization chambers with thin walls are occasionally used for personnel monitoring for beta radiation. Here, the reading is dependent upon the manner in which the chamber is worn; and care should be taken to see that it is not shielded by other devices or clothing. Some makes of ion chambers can be discharged by sudden shocks or by the presence of a leaky insulator, and it is advisable to wear such chambers in pairs in order to increase the probability of obtaining a correct exposure record.

b. Film Badges

In situations where exposure to mixed beta and gamma radiation is possible, or when the duration of exposure is relatively long, a film badge containing x-ray film is frequently used. In this badge, shields of various materials and thicknesses enable one to estimate the dose from different kinds and energies of radiations. Tin, silver, cadmium, or lead are commonly used for this purpose as well as to reduce the energy dependence of the film for gamma radiation. An estimate of the beta radiation is frequently made from the unshielded portion of the film after corrections are made for gamma-ray blackening and absorption by the film wrapping. Such an estimate is difficult to make when the energies of the radiation are not well defined, and the results can be regarded only as estimates in this case. Since most of the photographic emulsions are 15 to 20 times as sensitive to photons of 50 to 100 kev as to photons of 1 mev, great care must be taken in interpreting the readings obtained in areas where both high energy and low energy gamma radiation exists. In some cases, it is advisable to use shields of several different materials so that the differences in the absorption with atomic number may be used to estimate the fraction of each energy present.

In the development of the film, both blank film which has not been exposed to radiation and standard exposed films of the same emulsion shall be processed in each batch as controls. Developing temperatures shall be controlled, and constant agitation of the developer should be used. The films shall be provided with identifying markings produced by a suitable x-ray exposure, by punch marks, or by other suitable means of positive identification. When visual comparison with control films processed simultaneously indicates an exposure of more than one-tenth of the permissible value, the film density should be measured with a quantitative densitometer.

6.2 Portable Survey Instruments

Portable survey instruments are used to locate sources of radiation and to measure the intensities of radiation which will be received by workers. Instruments in which the response is proportional to the dose rate, such as ionization chambers, are useful for measurement; whereas those whose response varies with the energy of the radiation, including Geiger, proportional, and scintillation counters, are useful primarily as detecting devices. Detecting devices may be used for measurement when the energy of the radiation is known and the instrument is calibrated for the particular conditions encountered.

Since alpha radiation has a very limited range and will not penetrate from external sources to sensitive portions of the body, instruments are used only for detecting alpha-emitting contamination in the laboratory. These detectors include air proportional counters; scintillation counters; and in some cases ionization chambers with a thin window which will permit the alpha particles to penetrate to the sensitive portion of the chamber. In the latter case, the ionization chamber is usually calibrated in units of the numbers of alpha particles passing through the chamber or the response to a standard alpha source without correction for geometry or self-absorption. Geiger counters with thin mica windows may be used to survey for carbon-14 or sulfur-35 contamination. For beta-gamma detectors, Geiger, proportional, and scintillation counters are used. In general, these devices are more sensitive than ionization chambers and permit rapid localization of sources.

Portable survey meters for measurement of dose usually incorporate ionization chambers, although instruments which integrate the total energy loss from each particle in proportional counters or scintillation counters are sometimes used. For measuring beta radiation, the ionization chamber should have a thin window which will permit the passage of the beta radiation with little attenuation. For measurement of gamma radiation, the ionization chamber should have a wall of air-equivalent or tissue-equivalent [69] material of sufficient thickness to provide equilibrium with the secondary radiations generated in the wall.

Care should be taken to see that scatter from other portions of the instrument is minimized and that the leads from the chamber to the electronic circuit do not constitute, in themselves, an ionization chamber which will add to the meter reading.

Handbooks 48 [55], 51 [5], 73 [6], and 55 [44] discuss typical uses of portable survey instruments as well as limitations to be placed on these instruments.

6.3 Fixed Monitors

Radiation measuring instruments are sometimes used to give continuous recording of the dose rate at fixed locations. Visible or audible alarms may be connected to such devices to warn of unsafe conditions or unexpected contamination. Where the probability of clothing contamination is high, fixed monitors may be located around a doorway to detect contamination on the clothing of people passing through the doorway.

Where a large number of people are to be surveyed in a limited time during changes in shift, leaving work to eat, etc., a combination instrument to register contamination on the hands and shoes of personnel may be justified. Such instruments are available in various combinations to monitor shoe soles and both sides of the hands simultaneously for alpha and/or beta emitters.

6.4 Air Samplers

Air samplers are used to measure the quantities of radioactive materials in the air resulting from the handling of unconfined radioactive materials. Such equipment is vital in the safe handling of unconfined radioactive material in any quantities since air-borne contamination is one of the most common sources of internal deposition of radionuclides in humans. Portable, semi-fixed, and installed samplers are used; the latter often in conjunction with automatic radiation measuring instruments, recorders, and alarms. Since radioactive contamination may become air-borne as dusts, sprays, gases, or vapors, care should be taken that the sampling system selected will collect the particular form in which the material is present.

Filters, electrostatic precipitators or impactors of various designs can be used to collect dust or spray samples. Filter papers should be selected for their ability to provide high efficiency collection of the particle sizes encountered, combined with low self-absorption of the radiation. Samples of air containing gases or vapors may be collected in evacuated containers with the gas later transferred to an ionization chamber for direct measurement of the ionization. Liquid scrubbers can be used to collect vapors and can be adapted to discriminate between chemical forms of contaminants. For certain specific problems "getter beds" of activated charcoal, granulated chemicals, or powdered metals may be used. A useful technique is to draw the air through or into an ionization chamber where the activity of the contained radioactive material is measured directly. One or more pretreatment devices can refine the results to specific parameters of interest.

Both air samples representative of the hazard to personnel and general samples representing the condition in a working area are of value. Air samples representative of the hazard to personnel should be taken at the breathing zone near where the individual is working by instruments which can be moved from place to place. Fixed instruments are frequently used to give an indication of the general contamination levels throughout the entire area. In sampling for dusts or particulates, great care should be taken to insure that the air flow into the sampling nozzle is isokinetic with no particle size discrimination. [74].

Handbook 51 [5] contains general information on equipment used for air sampling plus recommendations for methods. Handbook 73 [6] gives specific tests for measuring and detecting a leakage from sealed sources.

6.5 Liquid Samples

Methods for sampling radioactive liquids depend upon such factors as the concentration of radionuclides in the solution, volume needed for analytical sensitivity, homogeneity of the solution, and accessibility for sampling.

As in air sampling, representative sampling of liquids is a requirement for proper evaluation. Static liquids should be well mixed at the time of sampling if possible. If not, then samples should be obtained from several locations in the container and either analyzed separately or combined proportionally for analysis. Flowing liquids may be sampled intermittently by bottle or dipper, weir and automatic scoop [75], solenoid valve tap or proportional pump [76], at a frequency dependent on expected fluctuations in flow and concentration. Depending on the desired interpretation of results, solids in the solution might be representatively sampled separately or included in the total analysis.

6.6 Air and Water Sample Counters

Laboratory instruments are frequently used to measure the quantity of radioactive materials in air and liquid samples. In order to compare such measurements to the maximum permissible concentrations for radiation protection, the results must be converted to units of concentration (preferably $\mu c/cc$). Due to the small amount of material required to absorb the alpha particles, alpha emitters are usually measured by means of counters so designed that the sample can be placed inside the chamber. The sample must be prepared in a thin layer or corrections must be applied for self-absorption in the sample. Such corrections are usually determined empirically for the particular absorbing material [77, 78].

Beta or gamma emitters may be measured with any sensitive counter or, in some cases by ionization chambers. Corrections for geometry, absorption and scatter may be determined separately or by measurement of a standard [79]. In measuring mixed alpha-beta contaminants, it should be noted that the alpha radiation will penetrate counter windows up to about eight or nine milligrams per square centimeter. Very weak beta emitters may be measured with windowless proportional counters or scintillation counters [80]. A gamma-ray spectrometer is an extremely powerful tool for measuring gamma emitters with high efficiency and distinguishing between radionuclides by means of the energy of the gamma radiation [81]. Such techniques will frequently minimize the amount of chemistry required in the analysis of samples. Calibration sources for the type and energy of radiation expected are primary tools in the measurement of radionuclides.

Special measurement techniques have been devised for measuring the quantity of radioactive material in gas and liquid samples. Other measuring techniques use photographic film, such as x-ray film for beta and gamma emitters, and nuclear track film for alpha emitters. Such techniques, although specialized, can frequently provide a solution to particular problems in the laboratory.

Handbooks 51 [5] and 80 [40] give additional information on the instrumentation and techniques required for the analysis of samples.

6.7 Calibrations

All instruments used in evaluating the radiation hazards in terms of exposure or absorbed dose, or for setting time limits of personnel irradiation shall be calibrated on a routine basis not less than once per quarter. Interim checks on the quantitative performance of the instrument should be made with small sources readily available to the user of the instrument and should include several points on the range. In general, reliability is to be valued above great precision. The user should be alert to detect instrumental breakdowns or errors in scale readings. If the instrument is to be used for a variety of nuclides, a complete calibration should include examination of the energy dependency by the use of two or more check points. The exposure rate and absorbed dose rate dependency should be measured up to the maximum rate for which the instrument is to be used. Battery checks should be made on the instrument at routine intervals depending upon the expected life of the batteries and the amount of use to which the instrument is put. Great care should be exercised in high radiation areas to avoid using instruments which may overload and thus read low or zero. Geiger counters are particularly susceptible to this fault, proportional counters and scintillation counters less so. Portable instruments should be checked for the presence of transferable contamination before allowing them to leave the work area for calibration or repair. Additional criteria for calibration, design, and maintenance of such instruments may be found in Handbooks 51 [5] and 80 [40].

7. Transportation of Radioactive Materials

Radioactive materials being transported are divorced from the controls effective where they are normally stored or used. Special precautions are, therefore, needed.

7.1 On-Site Transfers

In an organization large enough that a person other than the one using the material is assigned to transport it, written procedures should be followed. Other precautions appropriate to transfers within such an organization are given in this section.

Each container used in transporting radioactive materials shall be marked or tagged with the radiation symbol (section 4.6) to warn personnel approaching from all reasonable directions. It should be tagged with all necessary information to provide the reader with knowledge of the hazards such as radiation levels and handling precautions. It should be adequately sealed against leakage, and if fragile protected against breakage. No significant external contamination should be present. Two lines of defense against leakage should be considered for transfers that involve as much as one gram of plutonium or its equivalent in terms of the biological hazard.

Transportation between rooms or adjacent buildings where radiation monitoring coverage is readily available usually requires simple (but mandatory) precautions. Procedures should be established and followed in case of accident for keeping spectators away, shutting off ventilation if air-borne contamination is possible, and calling for aid (see section 5.9).

If the transportation requires an automobile, truck, or railroad car, beyond the immediate range of full radiation protection coverage, additional precautions are necessary. Each shipment should be accompanied by a form–"radioactive shipment record"–which describes the origin, destination, contents, radiation level at the outside of the container, contamination status of the outside of the container, precautions required during handling and transportation, exposure rates at locations in or adjacent to the vehicle where personnel exposure in excess of a few mrem may occur, and other information necessary for authorization, record, or inventory purposes. A sample form is given in appendix D.

For transportation of high levels of unsealed radionuclides by automobile or truck, two people should accompany the shipment. In some cases an escort vehicle is desirable. If the radiation level at points of occupancy in the vehicle during transportation would result in excessive exposure, plans should be adopted to substitute personnel during the shipment and suitable personnel meters should be provided.

The person in charge of the shipment should institute reasonable precautions against spilling of radioactive material from the container or conveyance. If a spill is detected, or in case of a fire or explosion, it is mandatory that the vehicle be stopped immediately. In such event, or if the vehicle fails in any other manner, one man should remain with the vehicle to keep other vehicles and persons away from the radioactive material until qualified help arrives. Repairs to the vehicle should require either keeping at a safe distance, continuous monitoring, or removal of any hazardous radiation source.

Persons involved with the shipment must take necessary precautions to prevent skin, internal, or clothing contamination. If personnel contamination is discovered, the persons involved shall report promptly for contamination survey, and for sampling of urine or feces if necessary.

7.2 Off-Site Transportation

For transportation of a radioactive source on public highways, it should be packaged in conformity with Interstate Commerce Commission regulations. In a private car, additional shielding may be required to protect all occupants of the car and persons who may approach the car casually or for servicing. The container should be carried as far as possible from occupants. It shall be marked with the name and address of the owner, the radiation symbol, a warning that the contents may be dangerous if removed, and a request that the owner be notified if the container is found. It may be advisable to lock the closed container.

If it is necessary to leave radioactive material in an unattended car, the container should be locked in the car, preferably in the luggage compartment. Any loss or theft of radioactive material that may constitute a potential public hazard shall be reported immediately to the local police or public health authorities.

The transportation of radioactive material by public carriers (truck, bus, railroad, airplane, boat) is subject to federal, state, and local regulations. Compliance with federal regulations ordinarily is satisfactory for state and local compliance. Federal regulations on this subject as of 1958 are given in reference [82]. For uniformity, the Interstate Commerce Commission regulations for railroad and trucks form the basis for regulations of the Civil Aeronautics Board, Coast Guard, and Post Office. Foreign regulations are also similar, except for notable differences in international air traffic requirements. Certain transportation companies also have unique restrictions resulting from license requirements or the make of their particular vehicles. Certain routes are barred because of tunnels, load limits, watersheds, etc. In arranging for transportation of radioactive material by public carriers, traffic agents should consult the applicable regulations as well as carrier representatives. Since these regulations are subject to change, the latest regulations should be consulted.

7.3 Transportation Containers

When glass bottles or flasks containing low- or medium-level radioactive solutions are to be carried between rooms or adjacent buildings, simple wooden, metal, or plastic carriers should be used to avoid breakage and to afford additional handling distance. With a handle spaced above the box a sufficient distance to keep hand exposures low, the low center of gravity will add insurance against spills.

The required thickness of shielding must be calculated to reduce exposure to the individual during handling along with his exposure from other sources to less than maximum permissible limits. The thickness may be calculated from data such as that given in appendix C. Openings through the shield must be stepped or offset to eliminate beams. The design should minimize exposure during loading and unloading.

Usually, lead is the most practical shielding material. For large containers requiring structural strength, the lead is usually cast into steel as a liner. Care is necessary to prevent voids; with a radiation source inside, voids may be detected by radiograph with film or survey with portable radiation instruments.

Handles should be located to bring the container close to the floor in the normal carrying position. Design of the container should encourage the use of the handles. Containers too heavy to carry may be mounted on wheels, with a handle for propulsion. Still larger containers should have hoisting rings, hooks, or trunnions.

Each transportation container should be marked with the radiation symbol and identification of the contents and the owner.

Disposable containers for waste often use concrete for shielding. Designs of all containers for offsite use require review, approval, and registration by the Bureau of Explosives unless they conform to specifications as provided in the tariffs.

8. Radioactive Waste Disposal

Radioactive wastes result from most operations with radionuclides, and proper handling requires that adequate attention be given to the disposal of these wastes so that human or other populations are not harmed. These wastes arise in a large variety of forms depending upon the particular use to which the radionuclides are put. For example, in chemical operations the wastes may be in the form of solutions, precipitates, contaminated equipment, contaminated containers, or agents used to decontaminate the laboratory. In medical or biological work the waste may consist of excreta or tissue specimens.

A convenient designation of wastes on a qualitative scale can be given by the use of the terms "high-level" and "low-level". In general high-level wastes are those which must be retained and disposed of by storage in an area where contact with humans or ecological populations is minimal. Low-level wastes can be then defined as those which can be disposed of to the environs of the laboratory by utilizing the natural dispersal phenomena available. A third category of intermediate wastes can often be concentrated for economical handling in the high-level category. It may be possible to use some property of the environment to concentrate and retain the radioactive materials in a region where contact with the ecological system is minimal. This is probably useful only in large installations where effective control can be maintained.

It should be noted that one cannot assure that all radioactive material is retained in the laboratory. There will be small quantities of radioactive wastes dispersed to the various waste systems in the laboratory even with the most stringent care. Even if complete analyses of air and water leaving the premises are available, the sensitivity of the analyses limits the extent of the knowledge of the quantity of radioactive material which escapes.

Radioactive nuclides cannot be destroyed except by natural decay. Thus decontamination of air and water can be achieved by deposition, absorption, or adsorption but not by destruction of the radionuclides. If the material settles out on the bed of a stream or in piping, it still retains its radioactivity even though the effluent stream appears to be inactive.

It should be noted that regulations concerning the disposal of radioactive wastes have been written by the AEC for those procuring isotopes on an AEC license [1], and that many states have laws or regulations which bear on waste disposal of these and other nuclides. The user should familiarize himself with the current regulations of the Federal, State, or Municipal government. Only general handling procedures for waste are included in this handbook. Recommendations for disposal of specific nuclides are given in other handbooks of this series [57, 58, 83].

8.1 Gaseous Wastes

Gaseous wastes can include radioactive materials in the form of gases, in the form of vapors, and in the form of small particles of solid materials which can be carried by air currents through the ventilation system to the environs. If large quantities of these materials are to be handled, the ventilation system should be equipped to remove the radionuclides (see section 4.2.c). For handling of solids under conditions where dust can be produced either directly by abrasive action or by evaporation of spray, high efficiency filters [52] should be provided in the exit air from hoods or other containment devices. For work with low levels of these nuclides as defined in table 2 the natural dilution from the ventilation air should be adequate to prevent overexposure (see section 4.2.c). It should be noted that with long use of a laboratory, however, there may be deposition and buildup of long-lived radioactive materials on the ductwork. This buildup can result in radiation dosages near the ductwork and possible sloughing of corrosion products from the internal surfaces of ducts to liberate radioactive particles to the environs. For handling gases or vapors special air cleaning equipment, depending upon the chemical or physical properties of the effluent, will be required.

The concentration of radioactive materials in the air leaving the ventilation system should not exceed the maximum permissible concentrations for breathing as given in Handbook 69 [17] unless regular and adequate monitoring or environmental surveys are carried out to prove the adequacy of the disposal system.

Where large quantities of radionuclides are routinely discharged to the environment, it is advisable to make environmental surveys in the vicinity since many radionuclides will be concentrated by absorption on surfaces. For example, the maintenance of an I-131 concentration in the atmosphere at the MPC given in Handbook 69 will result in deposition of the iodine on grasses and other crops. This iodine may be transferred to humans by ingestion of the vegetation or by drinking of milk from animals grazing on the contaminated pasture [24]. Such phenomena are dependent upon the nature of the gaseous wastes and the surrounding terrain and can be adequately controlled only by direct measurements.

Measurements of the quantity and identity of radionuclides discharged in gaseous wastes to the atmosphere should be recorded permanently. The accuracy required in such an estimate depends upon the quantities of radionuclides which are actually discharged to the environs. Additional information on disposal of gaseous wastes is given in National Bureau of Standards Handbooks 53 [58] and 61 [61].

8.2 Liquid Wastes

Liquid wastes may result from chemical operations, decontamination operations, or biological experiments. The objective of the liquid waste disposal program is to provide economical storage or dispersion of the wastes in such a manner as to insure safety of the surrounding populations.

Liquid wastes shall not be discharged to the sanitary system or other waste disposal system in such quantities that the radiation exposure of people exposed to such effluents exceeds the appropriate limit [84, 85]. If monitoring of the environment is not carried out, the quantities discharged should be such that the concentration at the point of discharge from the controlled area does not exceed the appropriate MPC for people in the environs. It may be noted, however, that a number of biological systems for concentrating certain radioactive materials exist. Thus, the slimes in sewage systems and the plants, fish, and fowl in streams may accumulate certain nuclides to a level greater than that of the water [21, 23, 86]. Such a reconcentration may result in significant dosage rates to the aquatic organisms or to individuals working on the sewage system. For this reason periodic surveys, with a frequency dependent upon the quantity of radionuclides discharged, should be made of the system into which radionuclides are placed. Handbook 49 [55] describes limits for the disposal of phorphorus-32 and iodine-131 and methods of diluting into a sewage system. Handbook 80 [40] provides an outline of methods to meet the requirements of CFR–10 Part 20 [1].

Small quantities of radionuclides which are too great for disposal into the existing sewage system should be stored in a shielded area and in unbreakable containers. If the half-life of the material is relatively short, such storage need be only until radioactive decay has lowered the quantity to a level where it can be dispersed to the environment. If the half-life is long, disposal should be by either permanent storage in a controlled area or disposal to the environment in a region where such disposal will not be hazardous to people or animals. Such disposal is rigidly controlled by the U.S. Atomic Energy Commission, and their recommendations must be followed. Several firms offer services for carrying out the disposal of such wastes.

8.3 Solid Wastes

Solid wastes may consist of chemical precipitates, contaminated equipment, materials used for decontamination, excreta, carcasses of animals, etc. Disposal methods consist of burial in the ground, sinking to the depths of the ocean, bulk reduction by incineration, or discarding in garbage or sewer. The method chosen depends upon the quantity of radioactive materials present in the wastes.

The waste materials which result from laboratory, medical, or industrial operations should be collected in suitable disposal containers conveniently located in each work area. Combustible materials may be segregated if an incinerator [87] with adequate air cleaning facilities is available. Frequent collections of the waste containers should be made; and if significant radiation dosage levels result from the containers, they should be removed from the laboratory in a special trip. If the solid material is contaminated with a short-lived radioactive material, it may be stored in a suitable area until the residual radioactivity is of insignificant hazard.

Extra care for radiation protection is necessary during the accumulation, collection, and disposal of radioactive solid wastes. Containers should be marked with the radiation symbol (see section 4.6) and any designations for segregation. Written procedures for safe handling during each stage will help to minimize contamination and personnel exposures. Frequent radiation monitoring should be emphasized and scheduled to detect carelessness or accidents. At all stages, containers of radioactive waste should be kept in controlled areas or under procedural control in transit between such areas. Special handling devices may be necessary for lifting or carrying containers which may cause high personnel exposure rates unless kept at a distance. "Hot spots" through the sides or bottom of the containers must not be overlooked.

Bulky contaminated equipment, such as machinery or process vessels, should be well sealed against liquid leakage, wrapped in waterproof paper or sheet plastic, and protected by crating to prevent tearing the wrapping during hoisting or dragging.

The irradiation to which the public might be subjected should not exceed the maximum permissible levels. Waste disposal must be controlled so that this criteria is met. Usually this can be accomplished by controlling the concentrations in air and water which may be consumed by the public. Methods of achieving such control for various wastes and radionuclides are discussed in detail in National Bureau of Standards Handbooks 49 [57], 53 [58], and 58 [83]. The principles discussed may be adapted to other wastes and radionuclides. Measurements or estimates of the quantity of radionuclides discarded to each site should be recorded.

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Appendix A

X- and Gamma Ray Attenuation Coefficients

The mass attenuation coefficients for several metals commonly used for shielding are given in table 7. Similar coefficients for water, concrete and air are given in table 8. These values were obtained from the tables given by G. White Grodstein [46].

Since an absorber attenuates a beam of X or gamma rays according to the amount of matter which the beam traverses, it is convenient to express the absorber thickness on a mass basis, in grams per square centimeter. Thus, the attenuation coefficient is expressed in cm^2/gm and is called the mass attenuation coefficient.

To convert these values to the linear coefficient used in the equation in section 4.1.c, multiply by the density of the material in grams/cubic centimeter.

The coefficients listed in the source [46] are given to three places. The numbers given in tables 7 and 8 were rounded off in some cases.

TABLE 7.—*Mass attenuation coefficient for metals commonly used in shields*

Photon energy Mev	Mass attenuation coefficient in cm^2/gm for—				
	Aluminum	Iron	Copper	Tungsten	Lead
0.01	26.3	178	224	58.1	80.1
.02	3.33	25.8	34.1	[a]52.6	[a]69.9
.04	.51	3.48	4.65	7.41	9.76
.06	.25	1.13	1.51	2.34	3.15
.08	.19	.56	.71	[a]7.49	1.41
.1	.16	.34	0.43	4.21	[a]5.29
.15	.13	.18	.21	1.44	1.84
.2	.12	.14	.15	.71	.90
.4	.092	.092	.092	.17	.21
.6	.078	.076	.075	.101	.114
.8	.068	.066	.065	.076	.084
1.0	.061	.060	.059	.064	.068
1.5	.050	.049	.048	.049	.051
2.0	.043	.042	.042	.044	.046
3.0	.035	.036	.036	.041	.042
4.0	.031	.033	.033	.040	.042
5.0	.028	.031	.032	.041	.043

[a] A resonance absorption in the bound electrons occurs at an energy somewhat lower than given.

TABLE 8.—*Mass attenuation coeficients for materials encountered in shielding problems*

Photon energy Mev	Mass attenuation coefficient in cm²/gm for—		
	Water	Concrete[a]	Air
0.01	5.10	24.6	4.89
.02	.72	3.34	.70
.04	.25	.54	.23
.06	.20	.27	.18
.08	.18	.20	.16
.1	.17	.17	.15
.15	.15	.14	.13
.2	.14	.12	.12
.4	.106	.095	.095
.6	.090	.080	.080
.8	.079	.071	.071
1.0	.071	.064	.064
1.5	.058	.052	.052
2.0	.049	.045	.045
3.0	.040	.036	.036
4.0	.034	.032	.031
5.0	.030	.029	.027

[a] Concrete varies in composition. Coefficients given are for a concrete with a density of 2.35 g/cm³ and composed of 0.56% H, 49.56% O, 31.35% Si, 4.56% Al, 8.26% Ca, 1.22% Fe, 0.24% Mg, 1.71% Na, 1.92% K, 0.12% S.

Appendix B

Buildup Factors

A correction is necessary in calculating the attenuation of gamma radiation through a shielding material because radiation is scattered in the shield in such a fashion that a portion of the scattered photons reach the observer. The correcting factor, called buildup factor is defined as the ratio of total dose from scattered plus unscattered photons to the dose from unscattered photons only. Its name derives from the fact that it increases to higher and higher values as the shield thickness increases. The buildup factor is included in calculations as:

$$I = BI_0e^{-\mu x}$$

where I_0 is the exposure rate of radiation before shielding:

I is the exposure rate after shielding:
B is the buildup factor;
μ is the absorption coefficient;
x is the shield thickness.

The buildup factor depends on the geometry of the source, shield, and observer. Factors are given below for point isotropic sources (table 9) and for plane monodirectional sources (beams) (table 10) for common shielding materials [47]. The accuracy of the results for high atomic number materials is estimated at roughly 5 percent for small penetrations, increasing to perhaps 15 percent at the largest penetration. The accuracy for low atomic number materials is not as good, being about 10 percent for medium penetrations and increasing to perhaps 30 percent for the greatest penetration.

The factors are given in terms of the dimensionless parameter of μx where μ is the attenuation coefficient and x is the shield thickness.

To illustrate the use of the tables, suppose a lead shield is required to attenuate the radiation from a Cobalt-60 source, from a level, unshielded, of 133 mR/hr at 1 meter, to a level of 10 mR/hr at 1 meter. For the average energy of 1.25 Mev, we find by interpolation in table 7, a mass absorption coefficient of about 0.059 cm^2/g. Multiplying by the density of lead, 11.4 g/cm^3, gives a linear absorption coefficient of 0.67 cm^{-1}. An attenuation factor of

13.3 corresponds to a μx value of 2.6. Interpolating in table 9, we find a buildup factor of about 1.9. Thus the attenuation must be calculated to include this factor

$$e^{\mu x} = \frac{BI_0}{I} = 1.9 \times 13.3 = 25,$$

or $\mu x = 3.3$. A second approximation would give 2.1 for the buildup, 28 for the attenuation, and 3.33 for μx. Dividing by the linear coefficient 0.67 cm^{-1}, we find 5 cm. of lead to be the required thickness. Without allowance for buildup we would have underestimated the thickness needed, as $2.6/0.67 = 4$ cm.

TABLE 9.—*Dose build-up factor for point isotropic source*

Photon energy Mev	μx						
	1	2	4	7	10	15	20
Water							
0.256	3.09	7.14	23.0	72.9	166	456	982
0.5	2.52	5.14	14.3	38.8	77.6	178	334
1	2.13	3.71	7.68	16.2	27.1	50.4	82.2
2	1.83	2.77	4.88	8.46	12.4	19.5	27.7
3	1.69	2.42	3.91	6.23	8.63	12.8	17.0
4	1.58	2.17	3.34	5.13	6.94	9.97	12.9
6	1.46	1.91	2.76	3.99	5.18	7.09	8.85
8	1.38	1.74	2.40	3.34	4.25	5.66	6.95
10	1.33	1.63	2.19	2.97	3.72	4.90	5.98
Aluminum							
0.5	2.37	4.24	9.47	21.5	38.9	80.8	141
1	2.02	3.31	6.57	13.1	21.2	37.0	58.5
2	1.75	2.61	4.62	8.05	11.9	18.7	26.3
3	1.64	2.32	3.78	6.14	8.65	13.0	17.7
4	1.53	2.08	3.22	5.01	6.88	10.1	13.4
6	1.42	1.85	2.70	4.06	5.49	7.97	10.4
8	1.34	1.68	2.37	3.45	4.58	6.56	8.52
10	1.28	1.55	2.12	3.01	3.96	5.63	7.32
Iron							
0.5	1.98	3.09	5.98	11.7	19.2	35.4	55.6
1	1.87	2.89	5.39	10.2	16.2	28.3	42.7
2	1.76	2.43	4.13	7.25	10.9	17.6	25.1
3	1.55	2.15	3.51	5.85	8.51	13.5	19.1
4	1.45	1.94	3.03	4.91	7.11	11.2	16.0
6	1.34	1.72	2.58	4.14	6.02	9.89	14.7
8	1.27	1.56	2.23	3.49	5.07	8.50	13.0
10	1.20	1.42	1.95	2.99	4.35	7.54	12.4
Lead							
0.5	1.24	1.42	1.69	2.00	2.27	2.65	2.73
1	1.37	1.69	2.26	3.02	3.74	4.81	5.86
2	1.39	1.76	2.51	3.66	4.84	6.87	9.00
3	1.34	1.68	2.43	3.75	5.30	8.44	12.3
4	1.27	1.56	2.25	3.61	5.44	9.80	16.3
5.1097	1.21	1.46	2.08	3.44	5.55	11.7	23.6
6	1.18	1.40	1.97	3.34	5.69	13.8	32.7
8	1.14	1.30	1.74	2.89	5.07	14.1	44.6
10	1.11	1.23	1.58	2.52	4.34	12.5	39.2

TABLE 10.—*Dose build-up factor for plane monodirectional source*

Photon energy Mev	μr					
	1	2	4	7	10	15
Iron						
0.5	2.07	2.94	4.87	8.31	12.4	20.6
1	1.92	2.74	4.57	7.81	11.6	18.9
2	1.69	2.35	3.76	6.11	8.78	13.7
3	1.58	2.13	3.32	5.28	7.41	11.4
4	1.48	1.90	2.95	4.61	6.46	9.92
6	1.35	1.71	2.48	3.81	5.35	8.30
8	1.27	1.55	2.17	3.27	4.58	7.33
10	1.22	1.44	1.95	2.89	4.07	6.70
Lead						
0.5	1.24	1.39	1.63	1.87	2.08	2.68
1	1.38	1.68	2.18	2.80	3.40	4.20
2	1.40	1.76	2.41	3.36	4.35	5.94
3	1.36	1.71	2.42	3.55	4.82	7.18
4	1.28	1.56	2.18	3.29	4.69	7.70
6	1.19	1.40	1.87	2.97	4.69	9.53
8	1.14	1.30	1.69	2.61	4.18	9.08
10	1.11	1.24	1.54	2.27	3.51	7.70

Appendix C

Nomograms for Shielding of Point Sources

Figures 2 and 3 in this appendix were prepared to simplify the calculation of shield thickness for lead and iron.

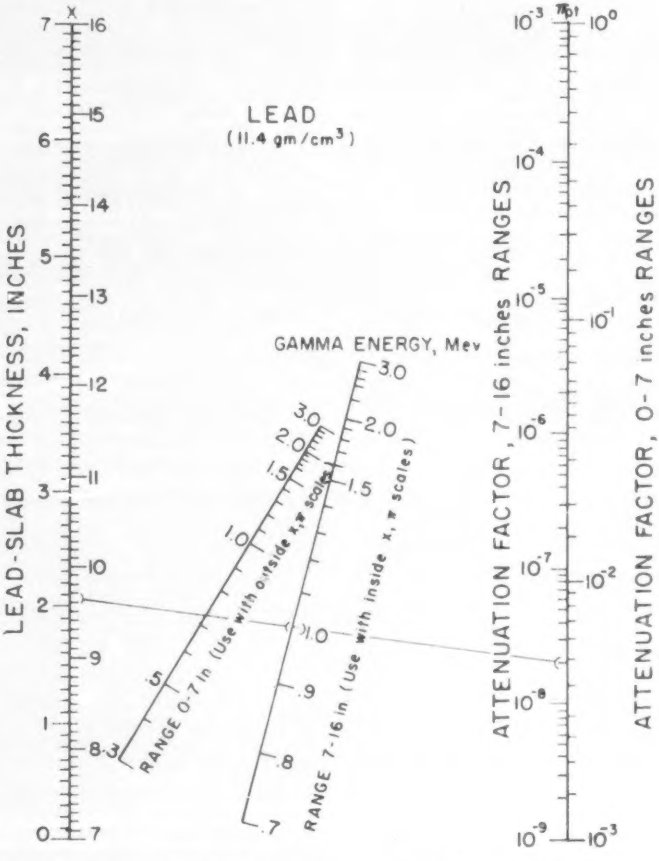

FIGURE 2. *Gamma attenuation with buildup in lead.*

(Chappell, D. G., Gamma attenuation with buildup in lead and iron, Nucleonics 15, No. 1, 52 (1957)).

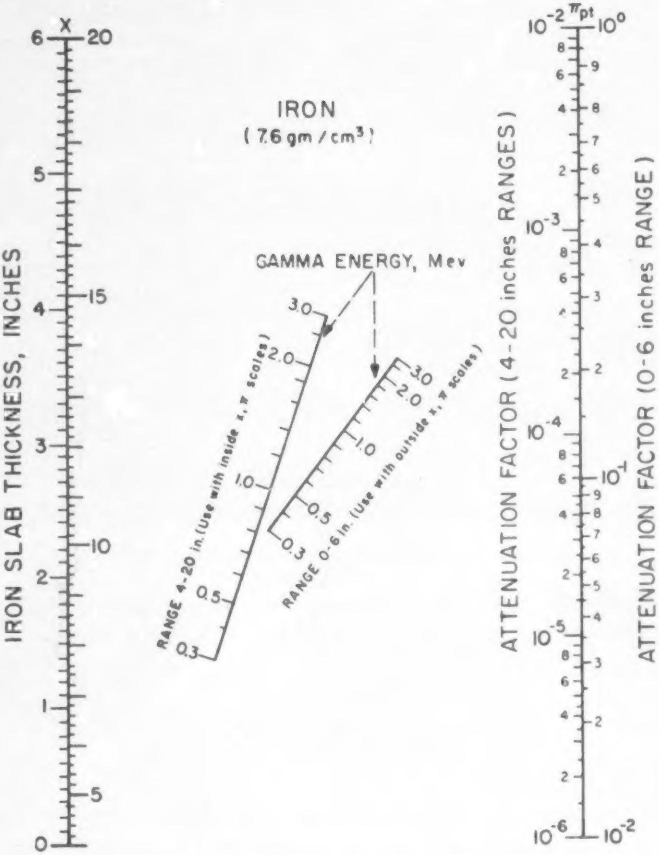

FIGURE 3. *Gamma attenuation with buildup in iron.*

(Chappell, D. G., Gamma attenuation with buildup in lead and iron, Nucleonics 15, No. 1, 52 (1957)).

They represent the attenuation for point sources of gamma emitters of various energies and include the scattered radiation. Over most of the range the accuracy is reported to be 10 percent or better, although at extreme thicknesses the accuracy is about 25 percent.

The example given in figure 2 answers the question as to how thick a lead shield must be to reduce the dose rate from a point source of 1.0 Mev gamma emitters by a factor of 2×10^{-8}. The line joins the desired attenuation factor and the gamma photon energy. The extrapolation of this line to the thickness scale indicates that the desired thickness is between 9.6 and 9.7 in.

Appendix D

Useful Forms

Forms 1 through 5 in this appendix are illustrative of those which have been found to be useful in radiation protection programs.

Throughout the personnel records, an individual is identified both by name and payroll number. In cases where a payroll number is not used or where the same numbers are reused upon termination of an employee, the individual's social security number makes a unique identification.

Form 1 may be useful for outlining a nonroutine work procedure. It provides space for indicating the type of work, survey readings, protective equipment, special personnel meters, and any special instructions such as actions following an accident, special surveys, etc. This form provides excellent communication between the person doing the work and the radiation safety officer as well as a record of the conditions encountered in various operations.

Form 2 can be used to record the results of a radiation survey.

Forms 3 and 4 are useful for recording decontamination procedures used. If such a form is made a part of the personnel record, it will provide information on localized exposure as well as a possible intake of radioactive materials.

Form 5 is a record of radioactive shipments. It provides a mechanism of informing the recipient of the nature and quantity of material involved as well as indicating any special instructions which should be followed. Such a form could also be used as a receipt to provide records on the location of radioactive materials.

FORM 1. *Form for recording radiation work procedure.*

RADIATION WORK PROCEDURE NO. REV.

VALID FROM TO

DESCRIPTION OF WORK

LOCATION:

JOB:

RADIATION CONDITIONS:

RADIATION MONITORING REQUIREMENTS:

SPECIAL INSTRUCTIONS:

PROTECTIVE EQUIPMENT REQUIREMENTS

HEAD
- ☐ CAP
- ☐ HOOD
- ☐ WATERPROOF HOOD

BODY
- ☐ NO PERSONAL OUTER CLOTHING
- ☐ LAB COAT
- ☐ ONE PAIR COVERALLS
- ☐ TWO PAIR COVERALLS
- ☐ WATERPROOF OUTER LAYER
- ☐

FEET
- ☐ SHOE COVERS
- ☐ CANVAS BOOTS
- ☐ RUBBERS
- ☐ BRITISH LEGGINGS
- ☐ HIP BOOTS

HANDS
- ☐ CANVAS GLOVES
- ☐ LEATHER GLOVES
- ☐ SURGEON'S GLOVES
- ☐ RUBBER GAUNTLET
- ☐ PLASTICIZED CANVAS
- ☐

RESPIRATORY
- ☐ RESPIRATORS
- ☐ ASSAULT MASKS
- ☐ HALF-MASKS
- ☐ CHEMOX MASKS
- ☐ FRESH AIR MASKS

METERS

☐ FILM BADGE
☐ GAMMA PENCILS
☐ NEUTRON FILM
☐ NEUTRON PENCILS
☐ X-RAY FILM
☐ SELF-READING PENCILS
☐ FINGER RINGS

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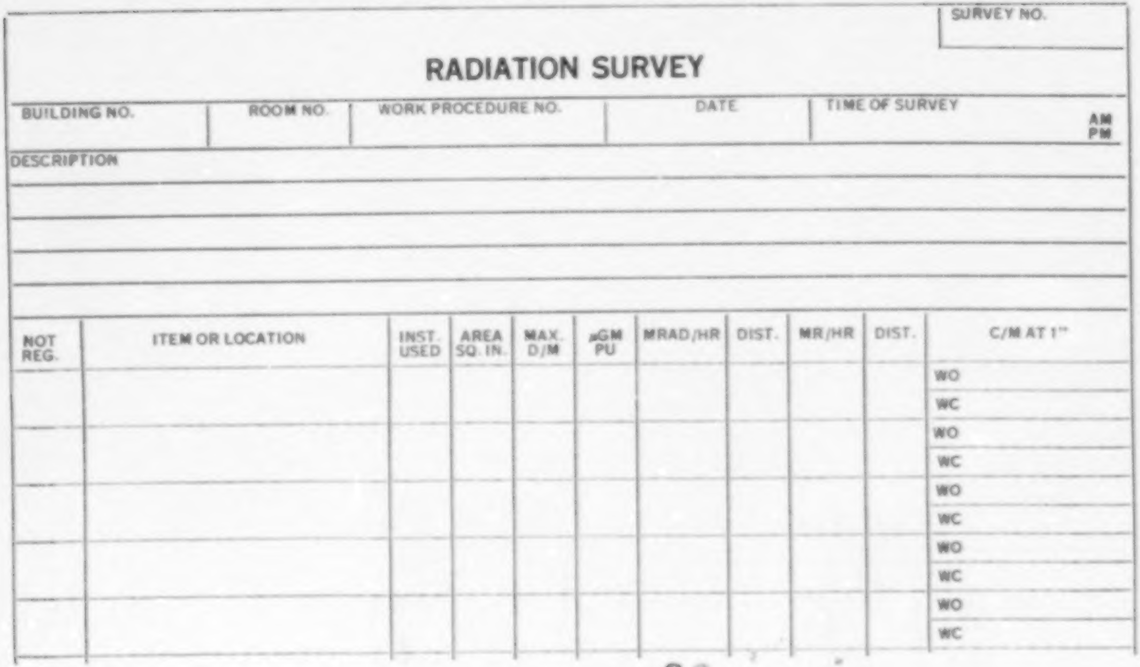
FORM 2. *Form for recording radiation survey.*



SURVEY NO.

RADIATION SURVEY

BUILDING NO.	ROOM NO.	WORK PROCEDURE NO.	DATE	TIME OF SURVEY AM PM

DESCRIPTION

NOT REG.	ITEM OR LOCATION	INST. USED	AREA SQ. IN.	MAX. D/M	µGM PU	MRAD/HR	DIST.	MR/HR	DIST.	C/M AT 1"
										WO WC
										WO WC
										WO WC
										WO WC
										WO WC

											WO WC
											WO WC
											WO WC
											WO WC
											WO WC

REMARKS:

MAXIMUM RADIATION MEASUREMENT OBTAINED:

MAXIMUM DOSE RATE

SIGNED

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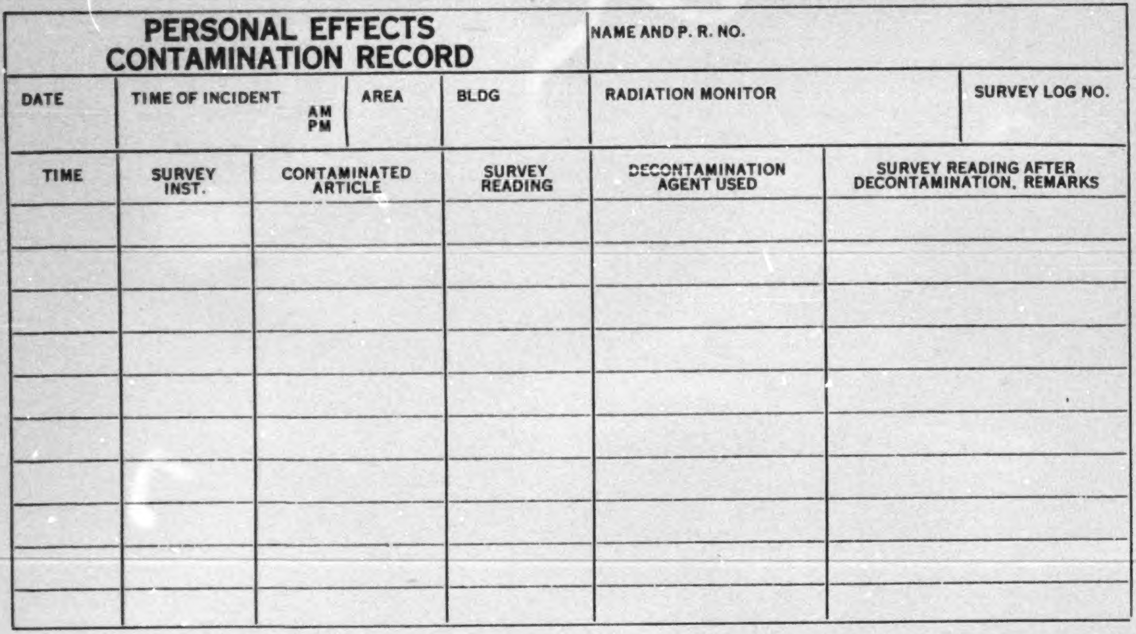
Form 3. *Form for recording personal effects contamination.*



PERSONAL EFFECTS CONTAMINATION RECORD

NAME AND P. R. NO.

DATE	TIME OF INCIDENT AM PM	AREA	BLDG	RADIATION MONITOR	SURVEY LOG NO.

TIME	SURVEY INST.	CONTAMINATED ARTICLE	SURVEY READING	DECONTAMINATION AGENT USED	SURVEY READING AFTER DECONTAMINATION, REMARKS

Form 4. *Form for recording skin decontamination.*

SKIN DECONTAMINATION RECORD	NAME AND P. R. NO.

DATE	TIME OF INCIDENT AM PM	AREA	BLDG.	RADIATION MONITOR	SURVEY LOG NO.

TIME	SURVEY INST.	SKIN AREA CONCERNED	SURVEY READING SHOW UNITS	DECONTAMINATION AGENT USED	SKIN CONDITION BEFORE AND AFTER TREATMENT, REMARKS, ETC.

Form 5. *Form for recording radioactive shipment.*

RADIOACTIVE SHIPMENT RECORD

☐ INTER - AREA
☐ OFF - PLANT

	NAME	AREA	UNIT	BLDG.	PHONE
FROM:					
TO:					

MONITORING RESULTS

CONTAINER	RADIATION LEVEL	AT DISTANCE	MATERIAL QUANTITY	RADIATION LEVEL	AT DISTANCE

MONITORED BY DATE

☐ NORMAL INSPECTION PERMITTED
☐ NO INSPECTION DUE TO RADIOACTIVITY

SPECIAL INSTRUCTIONS

1. SHIPMENT IS TO BE IDENTIFIED WITH TAGS, STICKERS, OR SIGNS.

2. RADIATION MONITORING IS TO BE NOTIFIED UPON ARRIVAL AT DESTINATION.

3. IT IS MANDATORY THAT THE CONVEYANCE BE STOPPED IMMEDIATELY AND RADIATION MONITORING AT THE ORIGIN NOTIFIED AS SOON AS POSSIBLE WHEN A SPILL IS DETECTED, THAT PRECAUTIONS BE TAKEN TO KEEP OTHER VEHICLES AND PERSONNEL AWAY FROM THE RADIOACTIVITY, AND THAT ONE MAN REMAIN WITH THE SHIPMENT UNTIL HELP ARRIVES. LIKEWISE, IF THE CONVEYANCE FAILS IN ANY MANNER SO THAT IT STOPS, ONE MAN SHALL REMAIN WITH THE SHIPMENT TO KEEP UNAUTHORIZED PERSONNEL AWAY.

4.

CHECKED OUT OF— AREA	BY	TIME	CHECKED IN— AREA	BY	TIME
		☐ AM ☐ PM			☐ AM ☐ PM

APPROVALS	ACKNOWLEDGEMENT OF RECEIPT	
RADIATION MONITORING	NAME	DATE
OPERATING UNIT (AUTHORIZED SIGNATORY, REF. RPS 1.1)	UNIT	

DISTRIBUTION: INTER-AREA SHIPMENT

WHITE : ACCOUNTABILITY REPRESENTATIVE
YELLOW : ACCOMPANY SHIPMENT—TO RECIPIENT
PINK : RADIATION MONITORING AT ORIGIN

OFF - PLANT SHIPMENT

1. ACCOUNTABILITY REPRESENTATIVE
2. RECIPIENT VIA MAIL
3. RADIATION MONITORING AT ORIGIN

Appendix E

Radiation Quantities and Units*

International Commission on Radiological Units and Measurements (ICRU) Report 10a 1962

1. Introduction

There has recently been much discussion of the fundamental concepts and quantities employed in radiation dosimetry. This has arisen partly from the rapid increase in the number of individuals using these concepts in the expanding field of nuclear science and technology, partly because of the need for extending the concepts so that they would be of use at higher photon energies and for particulate as well as for photon radiation, but chiefly because of certain obscurities in the existing formulation of the quantities and units themselves.

The roentgen, for example, was originally defined to provide the best quantitative measure of exposure to medium energy x radiation which the measuring techniques of that day (1928) permitted. The choice of air as a standard substance was not only convenient but, also appropriate for a physical quantity which was to be correlated with the biological effect of x rays, since the effective atomic number of air is not very different from that of tissue. Thus a given biological response could be reproduced approximately by an equal exposure in roentgens for x-ray energies available at that time. Since 1928 the definition of the roentgen has been changed several times, and this has reflected some feeling of dissatisfaction with the clarity of the concept.

The most serious source of confusion was the failure to define adequately the radiation quantity of which the roentgen was said to be the unit.[1] As a consequence of this omission the roentgen had gradually acquired a double role. The use of this name for the unit had become recognized as a way of specifying not only the magnitude but also the nature of the quantity measured. This practice conflicts with the general usage in physics, which permits, within the same field, the use of a particular unit for all quantities having the same dimensions.

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* Taken from Radiation Quantities and Units, International Commission on Radiological Units and Measurements, National Bureau of Standards Handbook 84, Washington, D.C. (1962).

[1] Franz, H. and Hubbner, W. Concepts and Measurement of Dose, Proceedings of Second International Conference on the Peaceful Uses of Atomic Energy, Geneva 1958, P/971 21, 101, United Nations, Geneva (1958).

Even before this, the need for accurate dosimetry of neutrons and of charged particles from accelerators or from radionuclides had compelled the International Commission on Radiological Units and Measurements (ICRU) to extend the number of concepts. It was also desired to introduce a new quantity which could be more directly correlated with the local biological and chemical effects of radiation. This quantity, *absorbed dose,* has a generality and simplicity which greatly facilitated its acceptance, and in a very few years it has become widely used in every branch of radiation dosimetry.

The introduction of absorbed dose into the medical and biological field was further assisted by defining a special unit–the *rad.* One rad is approximately equal to the absorbed dose delivered when soft tissue is exposed to one roentgen of medium voltage x radiation. Thus in many situations of interest to medical radiology, but not in all, the numbers of roentgens and rads associated with a particular medical or biological effect are approximately equal and experience with the earlier unit could be readily transferred to the new one. Although the *rad* is merely a convenient multiple of the fundamental unit, erg/g, it has already acquired, at least in some circles, the additional connotation that the only quantity which can be measured in rads is absorbed dose. On the other hand, the rad has been used by some authors as a unit for a quantity called by them *first collision dose;* this practice is deprecated by the Commission.

Being aware of the need for preventing the emergence of different interpretations of the same quantity, or the introduction of undesirable, unrelated quantities or units in this or similar fields of measurement, the ICRU set up, during its meeting in Geneva in September 1958, an *Ad Hoc* Committee. The task of this committee was to review the fundamental concepts, quantities, and units which are required in radiation dosimetry and to recommend a system of concepts and a set of definitions which would be as far as possible, internally consistent and of sufficient generality to cover present requirements and such future requirements as can be foreseen. The committee was instructed to pay more attention to consistency and rigor than to the historical development of the subject and was authorized to reject any existing quantities or units which seemed to hinder a consistent and unified formulation of the concepts.

Bertrand Russell[2] in commenting on the use and abuse of the concept of infinitesimals by mathematicians, remarks: "But mathematicians did not at first pay heed to (these) warnings. They went ahead and developed their science, and it is well that they should have done so. It is a peculiar fact about the genesis and growth of new disciplines that too much rigor too early imposed stifles the imagination and stultifies invention. A certain freedom from the strictures of sustained formality tends to promote the development of a subject in its early stages, even if this means the risk of a certain amount of error. Nonetheless, there comes a time in the development of any field when standards of rigor have to be tightened."

The purpose of the present reexamination of the concepts to be employed in radiation dosimetry was primarily "to tighten standards of rigor". If, in the process, some increased formality is required in the definitions in order to eliminate any foreseeable ambiguities, this must be accepted.

2. General Considerations

The development of the more unified presentations of quantities and units which is here proposed was stimulated and greatly assisted by mathematical models of the dosimetric field which had been proposed by some members of the committee in an effort to clarify the concepts. It appeared, however, that the essential features of the mathematical models had been incorporated into the definitions and hence the need for their exposition in this report largely disappeared. The mathematical approach is published elsewhere.[3]

As far as possible, the definitions of the various fundamental quantities given here conform to a common pattern. Complex quantities are defined in terms of the simpler quantities of which they are comprised.

The passage to a "macroscopic limit" which has to be used in defining point quantities in other fields of physics can be adapted to radiation quantities and a special discussion of this is included in the section headed "limiting procedures".

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[2] Russell, B., Wisdom of the West, p. 280 (Doubleday and Co., Inc., New York, 1959).

[3] Rossi, H. H. and Roesch, W. C., Field Equations in Dosimetry, Radiation Res. 16, 783 (1962).

The general pattern adopted is to give a short definition and to indicate the precise meaning of any special phrase or term used by means of an explanatory note following the definition. There has been no attempt to make the list of quantities which are defined here comprehensive. Rather, the Commission has striven to clarify the fundamental dosimetric quantities and a few others (such as activity) which were specifically referred to it for discussion.

It is recognized that certain terms for which definitions are proposed here are of interest in other fields of science and that they are already variously defined elsewhere. The precise wording of the definition and even the name and symbol given to any such quantity, may at some future date require alteration if discussions with representatives of the other interested groups of scientists should lead to agreement on a common definition or symbol. Although the definitions presented here represent some degree of compromise, they are believed to meet the requirements in the field of radiation dosimetry.

3. Quantities, Units, and Their Names

The Commission is of the opinion that the definition of concepts and quantities is a fundamental matter and that the choice of units is of less importance. Ambiguity can best be avoided if the defined quantity which is being measured is specified. Nevertheless, the special units do exist in this as in many other fields. For example, the hertz is restricted, by established convention, to the measurement of vibrational frequency, and the curie, in the present recommendations, to the measurement of the activity of a quantity of a nuclide. One does not measure activity in hertz nor frequency in curies although these quantities have the same dimensions.

It was necessary to decide whether or not to extend the use of the special dosimetric units to other more recently defined quantities having the same dimensions, to retain the existing restriction on their use to one quantity each, or to abandon the special units altogether. The Commission considers that the addition of further special units in the field of radiation dosimetry is undesirable, but continues to recognize the existing special units. It sees no objection, however, to the expression of any defined quantity in the appropriate units of a coherent physical system. Thus, to express absorbed dose in ergs per gram or joules per kilogram, exposure in coulombs per kilogram or activity in reciprocal seconds, are entirely acceptable alternatives to the use of the special units which, for historical reasons, are usually associated with these quantities.

The ICRU recommends that the use of each special unit be restricted to one quantity as follows:

The rad—solely for absorbed dose
The roentgen—solely for exposure
The curie—solely for activity.

It recommends further that those who prefer to express quantities such as absorbed dose and kerma (see below) in the same units should use units of an internationally agreed coherent system.

Several new names are proposed in the present report. When the absorbed dose concept was adopted in 1953, the Commission recognized the need for a term to distinguish it from the quantity of which the roentgen is the unit. In 1956 the Commission proposed the term *exposure* for this latter quantity. To meet objections by the ICRP, a compromise term, "exposure dose" was agreed upon.[4] While this term has come into some use since then, it has never been considered as completely satisfactory. In the meantime, the basic cause of the ICRP objection has largely disappeared since most legal codes use either the units rad or rem.

Since in this report the whole system of radiological quantities and units has come under critical review, it seemed appropriate to reconsider the 1956 decision. Numerous names were examined as a replacement for exposure dose, but there were serious objections to any which included the word dose. There appeared to be a minimum of objection to the name *exposure* and hence this term has been adopted by the Commission with the hope that the question has been permanently settled. It involves a minimum change from the older name exposure dose. Furthermore, the elimination of the term "dose" accomplishes the long-felt desire of the Commission to retain the term dose for one quantity only—the absorbed dose.

The term "*RBE* dose" has in past publications of the Commission not been included in the list of definitions but was merely presented as a "recognized symbol." In its 1959 report the Commission also expressed misgivings over the utilization of the same term, "*RBE*", in both radiobiology and radiation protection. It now recommends that the term *RBE* be used in radiobiology only and that another name be used for the linear-energy-transfer-dependent factor by which absorbed doses are to be multiplied to obtain for purposes of radiation protection a quantity that expresses on a common scale for all ionizing radiations the irradiation incurred by exposed persons. The name recommended for this factor is the *quality factor,* (*QF*). Provisions for other factors are also made. Thus a *distribution factor,* (*DF*), may be used to express the modification of biological effect due to non-uniform distribution of internally deposited isotopes. The product of absorbed dose and modifying factors is termed the *dose equivalent,* (*DE*). As a result of discussions between ICRU and ICRP the following formulation has been agreed upon:

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[4] For details see ICRU, 1956 Report, NBS Handb. 62, p. 2 (1957).

The Dose Equivalent

1. For protection purposes it is useful to define a quantity which will be termed the "dose equivalent", (*DE*).
2. (*DE*) is defined as the product of absorbed dose, *D*, quality factor, (*QF*), dose distribution factor, (*DF*), and other necessary modifying factors.

$$(DE)=D(QF)(DF)\ldots.$$

3. The unit of dose equivalent is the "rem". The dose equivalent is numerically equal to the dose in rads multiplied by the appropriate modifying factors.

Although this statement does not cover a number of theoretical aspects (in particular the physical dimensions of some of the quantities) it fulfills the immediate requirement for an unequivocal specification of a scale that may be used for numerical expression in radiation protection.

Another new name is that for the quantity which represents the kinetic energy transferred to charged particles by the uncharged particles per unit mass of the irradiated medium. This is the same as one of the common interpretations of a concept "first collision dose," that has proved to be of great value in the dosimetry of fast neutrons. The concept is also closely related to the energy equivalent of exposure in an x-ray beam. The name proposed, *kerma*, is based on the initials of *k*inetic *e*nergy *r*eleased in *ma*terial.

Still another new name is the *energy fluence* which is here attached to the quantity in the 1953 ICRU report called *quantity of radiation*. The latter term was dropped in the 1956 ICRU report but the concept—time integral of intensity—remains a useful one and the proposed term appears to be acceptable in other languages as well as English. A related quantity, *particle fluence,* which is equivalent to the quantity *nvt* used in neutron physics, is included to round out the system of radiation quantities.

The quantity for which the curie is the unit was referred to the committee for a name and definition. Hitherto the curie has been defined as a *quantity of the radioactive nuclide* such that 3.7×10^{10} disintegrations per second occur in it. However, it has never been specified what was meant by quantity of a nuclide, whether it be a number, mass, volume, etc. Meanwhile the custom has grown of identifying the number of curies of a radionuclide with its transformation rate. Because of the vagueness of the original concept, because of the custom of identifying curies with transformation rate and because it appeared not to interfere with any other use of the curie, the Commission recommends that the term *activity* be used for the transformation rate, and that the curie be made its unit. It is recognized that the definition of the curie is of interest to other bodies in addition to the ICRU, but by this report we recommend that steps be taken to redefine it as $3.7\times10^{10}s^{-1}$, i.e., as a unit of activity and not of quantity of a nuclide.

It is also recommended that the term *specific gamma ray constant* be used instead of *specific gamma ray emission* for the quotient of the exposure rate at a given distance by the activity. The former term focuses attention on the *constancy* of this quotient for a given nuclide rather than the *emission* of the source.

4. Detailed Considerations

A. Limiting Procedures

Except in the case of a uniform distribution of sources throughout a large region, radiation fields are in general non-uniform in space. They may also be variable in time. Many of the quantities defined in this report have to be specified as functions of space or time, and in principle they must therefore be determined for sufficiently small regions of space or intervals of time by some limiting procedure. There are conceptual difficulties in taking such limits for quantities which depend upon the discrete interactions between radiations and atoms. Similar difficulties arise with other macroscopic physical quantities such as density or temperature and they must be overcome by means of an appropriate averaging procedure.

To illustrate this procedure we may consider the measurement of the macroscopic quantity "absorbed dose" in a non-uniform radiation field. In measuring this dose the quotient of energy by mass must be taken in an elementary volume in the medium which on the one hand is so small that a further reduction in its size would not appreciably change the measured value of the quotient energy by mass and on the other hand is still large enough to contain many interactions and be traversed by many particles.[5] If it is impossible to find a mass such that both these conditions are met, the dose cannot be established directly in a single measurement. It can only be deduced from multiple measurements that involve extrapolation or averaging procedures. Similar considerations apply to some of the other concepts defined below. The symbol Δ precedes the symbols for quantities that may be concerned in such averaging procedures.

In the measurement of certain material constants such as stopping power, absorption coefficient, etc., the limiting procedure can be specified more rigorously. Such constants can be determined for a given material with any desired accuracy without difficulties from statistical fluctuations. In these cases the formulae quoted in the definitions are presented as differential quotients.

B. Spectral Distributions and Mean Values

In practice many of the quantities defined below to characterize a radiation field and its interaction with matter are used for radiations having a complex energy spectrum. An important general concept in this connection is the *spectral concentration* of one quantity with respect to another. The spectral concentration is the ordinate of the distribution function of the first quantity with respect to the second. The independent quantity need not always be energy or frequency; one can speak of the spectral concentration of flux density with respect to quantum energy or of the absorbed dose with respect to linear energy transfer. The interaction constants (such as μ, S and W) referred to in this report are often mean values taken over the appropriate spectral distributions of the corresponding quantities.

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[5] In interpreting radiation effects the macroscopic concept of absorbed dose may not be sufficient. Whenever the statistical fluctuations around the mean value are important, additional parameters describing the distribution of absorbed energy on a microscopic scale are necessary.

C. Units

For any of the quantities defined below the appropriate unit of an internationally agreed coherent system can be used. In addition certain special units are reserved for special quantities:

the rad for absorbed dose
the roentgen for exposure
the curie for activity.

D. Definitions

(1) *Directly ionizing particles* are charged particles (electrons, protons, α-particles, etc.) having sufficient kinetic energy to produce ionization by collision.

(2) *Indirectly ionizing particles* are uncharged particles (neutrons, photons, etc.) which can liberate directly ionizing particles or can initiate a nuclear transformation.

(3) *Ionizing radiation* is any radiation consisting of directly or indirectly ionizing particles or a mixture of both.

(4) The *energy imparted* by ionizing radiation to the matter in a volume is the difference between the sum of the energies of all the directly and indirectly ionizing particles which have entered the volume and the sum of the energies of all those which have left it, minus the energy equivalent of any increase in rest mass that took place in nuclear or elementary particle reactions within the volume.

NOTES: (a) The above definition is intended to be exactly equivalent to the previous meanings given by the ICRU to "energy retained by matter and made locally available" or "energy which appears as ionization, excitation, or changes of chemical bond energies". The present formulation specifies what energy is to be included without requiring a lengthy, and possibly incomplete, catalogue of the different types of energy transfer.

(b) Ultimately, most of the energy imparted will be degraded and appear as heat. Some of it, however, may appear as a change in interatomic bond energies. Moreover, during the degradation process the energy will diffuse and the distribution of heat produced may be different from the distribution of imparted energy. For these reasons the energy imparted cannot always be equated with the heat produced.

(c) The quantity *energy imparted to matter* in a given volume is identical with the quantity often called *integral absorbed dose* in that volume.

(5) The *absorbed dose* (D) is the quotient of ΔE_D by Δm, where ΔE_D is the energy imparted by ionizing radiation to the matter in a volume element, Δm is the mass of the matter in that volume element and Δ has the meaning indicated in section 4.A.

$$D=\frac{\Delta E_D}{\Delta m}$$

The special unit of absorbed dose is the *rad.*

$$1 \text{ rad}=100 \text{ erg}/g=\frac{1}{100} \text{ J}/kg$$

NOTE: J is the abbreviation for Joule

(6) The *absorbed dose rate* is the quotient of ΔD by Δt, where ΔD is the increment in absorbed dose in time Δt and Δ has the meaning indicated in section 4.A.

$$\text{Absorbed dose rate}=\frac{\Delta D}{\Delta t}$$

A special unit of absorbed dose rate is any quotient of the rad by a suitable unit of time (rad/*d*, rad/min, rad/*h*, etc.).

(7) The *particle fluence*[6] or *fluence* (Φ) of particles is the quotient of ΔN by Δa, where ΔN is the number of particles which enter a sphere[7] of cross-sectional area Δa and Δ has the meaning indicated in section 4.A.

$$\Phi=\frac{\Delta N}{\Delta a}$$

(8) The *particle flux density* or *flux density* (φ) of particles is the quotient of $\Delta\Phi$ by Δt where $\Delta\Phi$ is the particle fluence in time Δt and Δ has the meaning indicated in section 4.A.

$$\varphi=\frac{\Delta\Phi}{\Delta t}$$

NOTE: This quantity may also be referred to as particle fluence rate.

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[6] This quantity is the same as the quantity, *nvt*, commonly used in neutron physics.

[7] This quantity is sometimes defined with reference to a plane of area Δa, instead of a sphere of cross-sectional area Δa. The plane quantity is less useful for the present purposes and it will not be defined. The two quantities are equal for a unidirectional beam of particles perpendicularly incident upon the plane area.

(9) The *energy fluence* (F) of particles is the quotient of ΔE_F by Δa, where ΔE_F is the sum of the energies, exclusive of rest energies, of all the particles which enter a sphere[8] of cross-sectional area Δa and Δ has the meaning indicated in section 4.A.

$$F=\frac{\Delta E_F}{\Delta a}$$

(10) The *energy flux density* or *intensity* (I) is the quotient of ΔF by Δt where ΔF is the energy fluence in the time Δt and Δ has the meaning indicated in section 4.A.

$$I=\frac{\Delta F}{\Delta t}$$

NOTE: This quantity may also be referred to as energy fluence rate.

(11) The *kerma*[9] (K) is the quotient of ΔE_K by Δm, where ΔE_K is the sum of the initial kinetic energies of all the charged particles liberated by indirectly ionizing particles in a volume element of the specified material, Δm is the mass of the matter in that volume element and Δ has the meaning indicated in section 4.A.

$$K=\frac{\Delta E_K}{\Delta m}$$

NOTES: (*a*) Since ΔE_K is the sum of the initial kinetic energies of the charged particles liberated by the indirectly ionizing particles, it includes not only the kinetic energy these charged particles expend in collisions but also the energy they radiate in bremsstrahlung. The energy of any charged particles is also included when these are produced in secondary processes occurring within the volume element. Thus the energy of Auger electrons is part of ΔE_K.

(b) In actual measurements Δm should be so small that its introduction does not appreciably disturb the radiation field. This is particularly necessary if the medium for which kerma is determined is different from the ambient medium; if the disturbance is appreciable an appropriate correction must be applied.

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[8] See footnote 7.

[9] Various other methods of specifying a radiation field have been used, e.g., for a neutron source the "first collision dose" in a standard material at a specified point (see Introduction).

(c) It may often be convenient to refer to a value of kerma or of kerma rate for a specified material in free space or at a point inside a different material. In such a case the value will be that which would be obtained if a small quantity of the specified material were placed at the point of interest. It is, however, permissible to make a statement such as: "The kerma for air at the point P inside a water phantom is . . ." recognizing that this is a shorthand version of the fuller description given above.

(d) A fundamental physical description of a radiation field is the intensity (energy flux density) at all relevant points. For the purpose of dosimetry, however, it may be convenient to describe the field of indirectly ionizing particles in terms of the kerma rate for a specified material. A suitable material would be air for electromagnetic radiation of moderate energies, tissue for all radiations in medicine or biology, or any relevant material for studies of radiation effects.

Kerma can also be a useful quantity in dosimetry when charged particle equilibrium exists at the position and in the material of interest, and bremsstrahlung losses are negligible. It is then equal to the absorbed dose at that point. In beams of x or gamma rays or neutrons, whose energies are moderately high, transient charged-particle equilibrium can occur; in this condition the kerma is just slightly less than the absorbed dose. At very high energies the difference becomes appreciable. In general, if the range of directly ionizing particles becomes comparable with the mean free path of the indirectly ionizing particles, no equilibrium will exist.

(12) The *kerma rate* is the quotient of ΔK by Δt, where ΔK is the increment in kerma in time Δt and Δ has the meaning indicated in section 4.A.

(13) The *exposure* (X) is the quotient of ΔQ by Δm, where ΔQ is the sum of the electrical charges on all the ions of one sign produced in air when all the electrons (negatrons and positrons), liberated by photons in a volume element of air whose mass is Δm, are completely stopped in air and Δ has the meaning indicated in section 4.A.

$$X=\frac{\Delta Q}{\Delta m}$$

The special unit of exposure is the roentgen (R).

$$1R=2.58\times 10^{-4} C/kg$$ [10]

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[10] This unit is numerically identical with the old one defined as 1 e.s.u. of charge per .001293 gram of air. C is the abbreviation for coulomb.

Notes: (a) The words "charges on all the ions of one sign" should be interpreted in the mathematically absolute sense.

(b) The ionization arising from the absorption of bremsstrahlung emitted by the secondary electrons is not to be included in ΔQ. Except for this small difference, significant only at high energies, the exposure as defined above is the ionization equivalent of the kerma in air.

(c) With present techniques it is difficult to measure exposure when the photon energies involved lie above a few Mev or below a few kev.

(d) As in the case of kerma (4D(11), note (c)), it may often be convenient to refer to a value of exposure or of exposure rate in free space or at a point inside a material different from air. In such a case the value will be that which would be determined for a small quantity of air placed at the point of interest. It is, however, permissible to make a statement such as: "The exposure at the point P inside a water phantom is . . . ."

(14) The *exposure rate* is the quotient of ΔX by Δt, where ΔX is the increment in exposure in time Δt and Δ has the meaning indicated in section 4.A,

$$\text{Exposure rate} = \frac{\Delta X}{\Delta t}$$

A special unit of exposure rate is any quotient of the roentgen by a suitable unit of time (R/s, R/min, R/h, etc.).

(15) The *mass attenuation coefficient* $\left(\frac{\mu}{\rho}\right)$ of a material for indirectly ionizing particles is the quotient of dN by the product of ρ, N, and dl where N is the number of particles incident normally upon a layer of thickness dl and density ρ, and dN is the number of particles that experience interactions in this layer.

$$\frac{\mu}{\rho} = \frac{1}{\rho N} \frac{dN}{dl}$$

Notes: (a) The term "interactions" refers to processes whereby the energy or direction of the indirectly ionizing particles is altered.

(b) For x or gamma radiations

$$\frac{\mu}{\rho}=\frac{\tau}{\rho}+\frac{\sigma}{\rho}+\frac{\sigma_{\mathrm{coh}}}{\rho}+\frac{\kappa}{\rho}$$

where $\frac{\tau}{\rho}$ is the mass photoelectric attenuation coefficient, $\frac{\sigma}{\rho}$ is the total Compton mass attenuation coefficient, $\frac{\sigma_{\mathrm{coh}}}{\rho}$ is the mass attenuation coefficient for coherent scattering, and $\frac{\kappa}{\rho}$ is the pair-production mass attenuation coefficient.

(16) The *mass energy transfer coefficient* $\left(\frac{\mu_K}{\rho}\right)$ of a material for indirectly ionizing particles is the quotient of dE_K by the product of E, ρ and dl where E is the sum of the energies (excluding rest energies) of the indirectly ionizing particles incident normally upon a layer of thickness dl and density ρ, and dE_K is the sum of the kinetic energies of all the charged particles liberated in this layer.

$$\frac{\mu_K}{\rho}=\frac{1}{E\rho}\frac{dE_K}{dl}$$

NOTES: (a) The relation between fluence and kerma may be written as

$$K=F\,\frac{\mu_K}{\rho}$$

(b) For x or gamma rays of energy $h\nu$

$$\frac{\mu_K}{\rho}=\frac{\tau_a}{\rho}+\frac{\sigma_a}{\rho}+\frac{\kappa_a}{\rho}$$

where

$$\frac{\tau_a}{\rho}=\frac{\tau}{\rho}\left(1-\frac{\delta}{h\nu}\right)$$

$\left(\frac{\tau}{\rho}\right.$ = the photoelectric mass attenuation coefficient, δ = average energy emitted as fluorescent radiation per photon absorbed.$\left.\right)$

and
$$\frac{\sigma_e}{\rho}=\frac{\sigma}{\rho}\frac{E_e}{h\nu}$$

$\left(\frac{\sigma}{\rho}\right.$ = total Compton mass attenuation coefficient, E_e = average energy of the Compton electrons per scattered photon.$\left.\right)$

and

$$\frac{\kappa_a}{\rho}=\frac{\kappa}{\rho}\left(1-\frac{2mc^2}{h\nu}\right)$$

$\left(\frac{\kappa}{\rho}\right.$ = mass attenuation coefficient for pair production, mc^2 = rest energy of the electron.

(17) The *mass energy-absorption coefficient* $\left(\frac{\mu_{en}}{\rho}\right)$ of a material for indirectly ionizing particles is $\frac{\mu_K}{\rho}$ $(1-G)$ where G is the proportion of the energy of secondary charged particles that is lost to bremsstrahlung in the material.

NOTES: (a) When the material is air, $\left(\frac{\mu_{en}}{\rho}\right)$ is proportional to the quotient of exposure by fluence.

(b) $\frac{\mu_K}{\rho}$ and $\frac{\mu_{en}}{\rho}$ do not differ appreciably unless the kinetic energies of the secondary particles are comparable with or larger than their rest energy.

(18) The *mass stopping power* $\left(\frac{S}{\rho}\right)$ of a material for charged particles is the quotient of dE_s by the product of dl and ρ, where dE_s is the average energy lost by a charged particle of specified energy in traversing a path length dl, and ρ is the density of the medium.

$$\frac{S}{\rho}=\frac{1}{\rho}\frac{dE_s}{dl}$$

NOTE: dE_s denotes energy lost due to ionization, electronic excitation and radiation. For some purposes it is desirable to consider stopping power with the exclusion of bremsstrahlung losses. In this case $\frac{S}{\rho}$ must be multiplied by an appropriate factor that is less than unity.

(19) The *linear energy transfer* (L) of charged particles in a medium is the quotient of dE_L by dl where dE_L is the average energy locally imparted to the medium by a charged particle of specified energy in traversing a distance of dl.

$$L=\frac{dE_L}{dl}$$

NOTES: (a) The term "locally imparted" may refer either to a maximum distance from the track or to a maximum value of discrete energy loss by the particle beyond which losses are no longer considered as local. In either case the limits chosen should be specified.

(b) The concept of linear energy transfer is different from that of stopping power. The former refers to energy imparted within a limited volume, the latter to loss of energy regardless of where this energy is absorbed.

(20) The *average energy* (W) *expended in a gas per ion pair formed* is the quotient of E by N_W, where N_W is the average number of ion pairs formed when a charged particle of initial energy E is completely stopped by the gas.

$$W=\frac{E}{N_W}$$

NOTES: (a) The ions arising from the absorption of bremsstrahlung emitted by the charged particles are not to be counted in N_W.

(b) In certain cases it may be necessary to consider the variation in W along the path of the particle, and a differential concept is then required, but is not specifically defined here.

(21) A *nuclide* is a species of atom having specified numbers of neutrons and protons in its nucleus.

(22) The *activity* (A) of a quantity of a radioactive nuclide is the quotient of ΔN by Δt where ΔN is the number of nuclear transformations which occur in this quantity in time Δt and Δ has the meaning indicated in section 4.A.

$$A=\frac{\Delta N}{\Delta t}$$

The special unit of activity is the curie (c).

$$1c=3.7\times10^{10}s^{-1} \text{ (exactly)}$$

Note: In accordance with the former definition of the curie as a unit of quantity of a radioactive nuclide, it was customary and correct to say: "Y curies of P–32 were administered . . . . ." It is still permissible to make such statements rather than use the longer form which is now correct: "A quantity of P–32 was administered whose activity was Y curies."

(23) The *specific gamma ray constant* (Γ) of a gamma-emitting nuclide is the quotient of $l^2 \frac{\Delta X}{\Delta t}$ by A, where $\frac{\Delta X}{\Delta t}$ is the exposure rate at a distance l from a point source of this nuclide having an activity A and Δ has the meaning indicated in section 4.A.

$$\Gamma=\frac{l^2\Delta X}{A\Delta t}$$

Special units of specific gamma ray constant are $Rm^2h^{-1}c^{-1}$ or any convenient multiple of this.

Note: It is assumed that the attenuation in the source and along l is negligible. However, in the case of radium the value of Γ is determined for a filter thickness of 0.5 mm of platinum and in this case the special units are $Rm^2h^{-1}g^{-1}$ or any conventient multiple of this.

TABLE 4.1.—*Table of quantities and units*

No.	Name	Symbol	Dimensions [a]	Units		
				MKSA	cgs	Special
4	Energy imparted (integral absorbed dose)	......	E	J	erg	g. rad.
5	Absorbed dose	D	EM^{-1}	J kg^{1-}	erg g^{-1}	rad.
6	Absorbed dose rate	......	$EM^{-1}T^{-1}$	J kg^{-1}s^{-1}	erg g^{-1}s^{1-}	rad s^{-1} etc.
7	Particle fluence or fluence	Φ	L^{-2}	m^{-2}	cm^{-2}	
8	Particle flux density	Φ	$L^{-2}T^{-1}$	m^{-2}s^{-1}	cm^{-2}s^{-1}	
9	Energy fluence	F	EL^{-2}	J m^{-2}	erg cm^{-2}	
10	Energy flux density or intensity.	I	$EL^{-2}T^{-1}$	J m^{-2}s^{-1}	erg cm^{-2}s^{-1}	
11	Kerma	K	EM^{-1}	J kg^{-1}	erg g^{-1}	
12	Kerma rate	......	$EM^{-1}T^{-1}$	J kg^{-1}s^{1}1	erg g^{-1}s^{-1}	
13	Exposure	X	QM^{-1}	C kg^{-1}	esu g^{-1}	R (roentgen).
14	Exposure rate	......	$QM^{-1}T^{-1}$	C kg^{-1}s^{-1}	esu g^{-1}s^{-1}	Rs^{-1} etc
15	Mass attenuation coefficient.	$\frac{\mu}{\rho}$	L^2M^{-1}	m^{2}kg^{-1}	cm^{2}g^{-}1	
16	Mass energy transfer coefficient.	$\frac{\mu k}{\rho}$	L^2M^{-1}	m^{2}kg^{-1}	cm^{2}g^{-1}	
17	Mass energy absorption coefficient.	$\frac{\mu_{en}}{\rho}$	L^2M^{-1}	m^{2}kg^{-1}	cm^{2}g^{-1}	
18	Mass stopping power	$\frac{S}{\rho}$	EL^2M^{-1}	J m^{2}kg^{-1}	erg cm^{2}g^{-1}	
19	Linear energy transfer	L	EL^{-1}	J m^{-1}	erg cm^{-1}	kev (μm)$^{-1}$
20	Average energy per ion pair.	W	E	J	erg	ev.
22	Activity	A	T^{-1}	s^{-1}	s^{-1}	c(curie).
23	Specific gamma-ray constant.	Γ	QL^2M^{-1}	Cm^{2}kg^{-1}	esu cm^{2}g^{-1}	Rm^{2}h^{-1}c^{-1} etc.
	Dose equivalent	DE	......	......	......	rem

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[a] It was desired to present only one set of dimensions for each quantity, a set that would be suitable in both the MKSA and electrostatic-cgs systems. To do this it was necessary to use a dimension Q, for the electrical charge, that is not a fundamental dimension in either system. In the MKSA system (fundamental dimensions M, L, T, I) Q represents the product IT; in the electrostatic-cgs system (M, L, T) it represents $M^{1/2}L^{3/2}T^{-1}$.

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